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NRC CEOG Project Number 692

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Subject: Submittal of Topical Report CE NPSD-683, Rev. 05, "The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical Specifications," July 2000

References:
1) Letter, S. K. Gambhir (OPPD) to USNRC Document Control Desk, "Application for Amendment of Operating License", LIC-99-0045, May 26, 1999
2) W. G. Gates (OPPD) to USNRC Document Control Desk, "Withdrawal of Application for Amendment of Operating License", LIC-99-0072, August 6, 1999

This letter submits CE Owners Group Report CE NPSD-683 Rev. 05 for Nuclear Regulatory Commission (NRC) review and approval. Enclosed for your use are eight (8) copies of the report, with an additional four (4) copies forwarded to J. S. Cushing. Revision 05 supercedes, in their entirety, all previous versions of this topical report (Note - Rev. 04 was prepared for internal purposes and was not submitted for NRC review). The last submittal of this topical report was Rev. 03 and was made by the Omaha Public Power District (OPPD) for its Fort Calhoun Station (Reference 1). OPPD withdrew Rev. 03 from NRC consideration on August 6, 1999 (Reference 2).

The enclosed report presents an approach for 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.

CE NPSD-683, Rev. 05 has been reformatted to conform with the guidance provided by the NRC in GL-96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits". Other changes to Rev. 05 reflect

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feedback obtained from NRC staff at various meetings and during telephone conference calls. Incorporation of this feedback included the expansion of the methodology discussion (Appendix D). Appendices B and C contain Technical Specification (TS) markups indicating the manner in which CE NPSD-683, Rev. 05 could be implemented by licensees. The appendices, however, are provided only as examples of implementation approaches and are not being provided for NRC review and approval. Necessary changes to plant specific TS or to NUREG-1432 (CE Standard TS) will be submitted under existing TS change processes.

OPPD, as CEOG lead plant, will submit a letter docketing their intent to reference CE NPSD-683, Rev. 05 and the PTLR methodology documented therein for their Fort Calhoun Station.

The NRC should address technical questions related to CE NPSD-683 Rev. 05 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

cc: J. Cushing, NRC w/4 copies
K. Holthaus, OPPD w/1 copy
CEOG Library (Task 1174)

CEOG COMBUSTION ENGINEERING OWNERS GROUP

CE NPSD-683
Rev. 05

DEVELOPMENT OF A RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR THE REMOVAL OF P-T LIMITS AND LTOP REQUIREMENTS FROM THE TECHNICAL SPECIFICATIONS

FINAL REPORT

CEOG TASK 1174

Prepared for the
C-E OWNERS GROUP
July 2000

Legal Notice

CE NPSD-683, REV. 05

THE DEVELOPMENT OF A RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR THE REMOVAL OF P-T LIMITS AND LTOP SETPOINTS FROM THE TECHNICAL SPECIFICATIONS

JULY 2000

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ACRONYMS

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

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

ABSTRACT

An approach is presented in this report for CEOG utilities to relocate the Pressure-Temperature (P-T) limit curves, low temperature overpressure protection (LTOP) setpoint curves and values currently contained in the Technical Specifications (TSs) to a licensee-controlled document. The approach is based upon criteria specified 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 requirements of GL 96-03 and contains the detailed information needed to support the pertinent LCOs, which would remain in the TS. This topical report contains current methodology descriptions of RCS P-T limit development, LTOP analyses, ART calculation, RV Surveillance Program and Calculation of Neutron Fluence, which support the PTLR. An example of a PTLR is prepared along with the proposed changes to the subject TS.

The enclosed sample PTLR is generic in nature and can be easily tailored to be suitable to any Combustion Engineering Nuclear Steam Supply System (CE NSSS) design.

This document has been prepared by CE Nuclear Power LLC, a subsidiary of Westinghouse Electric Company LLC ("Westinghouse") for the Combustion Engineering Owners Group ("CEOG"). The methods presented are applicable to Combustion Engineering Nuclear Steam Supply System (CE NSSS) designs.

A INTRODUCTION

In an effort to improve the maintenance of Technical Specifications (TSs), the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 96-03 (Reference 3), which allows the relocation of requirements from the TSs into another licensee controlled document called a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR). This relocation enhances the regulatory processing of frequently revised items such as RCS Pressure-Temperature (P-T) limits, Low Temperature Overpressure Protection (LTOP) setpoints, 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 requirements of GL 96-03. The GL requires 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 that the licensee submits a license amendment request to obtain NRC approval of the proposed methodology prior to implementing its use in a PTLR.

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 4-P of CE NPSD-683-P is a total revision and supercedes all previous revisions (i.e., Revisions 0 through 3).

A.1 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" was published in the Federal Register (July 17, 1973) and 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 currently operating and under construction. 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.

A.2 DESCRIPTION OF ACTIVITIES

The NRC issued GL 96-03 to advise 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 controlled document. This topical report addresses the required information to be included in the PTLR based on the GL 96-03. The guidance is divided into seven provisions to be addressed in the PTLR. They 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

Each provision requires a methodology description be provided along with specific data about the operating plant. These provisions are specifically addressed in sections 1-7 of this topical report in conformance with the matrix of GL 96-03. The example PTLR provided in Appendix A is organized to address each provision.

Appendices B & C provide example markups of the changes necessary to integrate the PTLR into individual plant technical specifications. Appendix B is an example of markups to Standard Technical Specifications and Appendix C is an example of markups for plants using the CE Improved Standard Technical Specification format. These markups are provided for information only and are not intended for formal NRC review.

A.2.1 PTLR DEVELOPMENT

The PTLR was developed on a generic basis so that it would apply to all utilities owning CE NSSS designed pressurized water reactors. A review of typical LCOs for RCS P-T limits and LTOP requirements was performed and included in the generic PTLR.

In order to support the PTLR, methodology descriptions were prepared and incorporated as Sections 1-7 herein. The methodologies presented describe the development of RCS P-T limits, LTOP setpoints, RV Surveillance Programs and neutron fluence values.

A.3 GENERIC PTLR

To facilitate development of a plant specific PTLR, an example PTLR is presented in Appendix A of this report. This sample PTLR is only a guide since each plant specific PTLR will be different depending on the many different plant specific features such as RCS materials, the type of valves used for LTOP, whether Code case N-640 is used, etc.

A.4 REACTOR COOLANT PRESSURE BOUNDARY OPERATIONAL DESCRIPTION

A.4.1 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 which include RV boltup, heatup and cooldown
2. Inservice hydrostatic pressure tests and leak tests
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 the particular operation.

A.4.2 NORMAL OPERATION

A.4.2.1 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.

A.4.2.2 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 net positive suction head (NPSH) or the RCP seal requirements, 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.

A.4.2.3 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 which 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 NPSH requirements. The cooldown rate must not exceed the appropriate rates specified by the P-T limits.

A.4.3 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 determined by the requirements of ASME Code Section XI (Table IWB-5222-1). The minimum temperature for the required pressure is determined by the fracture toughness requirements and guidance provided in 10 CFR 50, Appendix G.

A.4.4 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 requirements of a proposed NRC Regulatory Guide (currently 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. The 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 was 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 requires 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.

To demonstrate the accuracy of the methodology, an analytic uncertainty analysis must be performed. 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).

This list is not necessarily exhaustive and 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.

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 were 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, the proposed modifications may be done using the 10CRF50.59 process if the proposed changes are consistent with the licensee's ASTM E-185 procedure of record, or with one of the more recent editions of the Standard Procedure listed in the rule (i.e., E185-73, E185-79, or E185-82).

[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 and outside in the annulus between the vessel and the biological shield.

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 requested and then granted by the Director, Office of Nuclear Regulation). 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 a value of the initial reference temperature, RT_{NDT}, and of 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. Some CEOG sponsored efforts which are pertinent are topical report CEN-189, December 1981, "Evaluation of Pressurized Thermal Shock Effects due to Small Break LOCAs with Loss of Feedwater for the Combustion Engineering NSSS", CE NPSD-1039, Revision 02, June 1997, CEOG Task 902, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds", and CE NPSD-1119, Revision 1, July 1998, CEOG Task 1054, "Updated Analysis for Combustion Engineering Fabricated Reactor Vessel Welds Best Estimate Copper and Nickel Contents."

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 limits developed using NRC-approved methodologies and contained in TSs 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.

As agreed upon with the NRC (Reference 21, pg. 1), no other methodologies beyond those currently used for CE NSSS designs are included herein, since anytime a licensee changes methodology, a license amendment is required.

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 paragraph (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) and another, 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 minimized and very low.

Consequently, the NRC has provided guidance to ensure overpressure protection for normal operation and 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, BTP RSB 5-2 requires protection of the Appendix G limits 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, developed via the ASME Code Case N-640 (Reference 11) methodology (with prior NRC approval), 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 and the RCS has pressure boundary integrity. The RCS does not have pressure boundary integrity when it is open to containment with a minimum area of the opening greater than, or equal to, a value specified in TS for a vent. The vent must be capable of mitigating the limiting LTOP events and the vent area must be justified by analysis. The LTOP enable temperature is a temperature under which the LTOP relief valves must be aligned to the RCS for automatic protection.

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.

The LTOP-related limitations are usually contained in TSs, 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 change, the TSs may need to be changed to incorporate these new P-T limits and/or LTOP requirements. The TS LTOP requirements may also be affected by plant modifications; if these adversely impact LTOP analyses.

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 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 TSs that contain the P-T limits, LTOP setpoints and limits, and LTOP enable temperatures can be controlled in the PTLR, rather than in the TSs. 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 TSs. 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 TSs 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 the initial NRC approval of this topical report and any 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. 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. NRC guidance on performing LTOP related analysis is documented in BTP 5-2 (Reference 12). It is important to refer to BTP RSB 5-2 while doing LTOP-related analyses.

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 of 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. LTOP evaluations may have different objectives, depending on:

   a) Whether a new LTOP system is designed, or

   b) The current LTOP limitations need to be verified due to new P-T limits and/or revised pressure transient analysis, or

   c) The existing limitations need to be relaxed.

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 TSs. 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 TSs. 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 the TSs) 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 TSs. To take credit for a restriction for transient mitigation in the pressure transient analyses, this restriction must be in the TSs. 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 are selected from the existing, or preliminary, PORV setpoint vs. temperature curve, from the lowest setpoint at the boltup temperature to the highest setpoint at the LTOP enable temperature, at 50 – 200 psi increments. The smaller setpoint increments must be used at the lower temperatures to provide more peak pressure data points where the operating window is most restricted.

Similarly, if several HPSI and/or charging pump combinations may be operable within the LTOP temperature range, each must be analyzed with each applicable setpoint and water-solid conditions.

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 required by 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.

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

  This assumption can be used if the TSs ensure operation with a steam volume. 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).

- *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 a single relief valve or Pressurizer steam volume.

- *A water-solid Pressurizer must be assumed, with water at saturation at the initial pressure.*

  This assumption must be used for bounding analyses. The assumption expedites the transient response and reduces discharge flow rates for the

PORVs and relief valves on the Pressurizer, as it reduces water subcooling at the inlet. If analyses are performed for other conditions as well, less limiting fluid conditions in the Pressurizer may be justified, such as subcooled water or steam. Subcooled water in the Pressurizer exists at the low reactor coolant temperatures, when the Pressurizer is filled but is not at saturation. A steam volume in the Pressurizer can be assumed if the TSs contain a limitation on a maximum Pressurizer water level or volume (or minimum steam volume) requirement for the LTOP temperature region, or a portion thereof.

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

  This input increases transient pressure.

- *Decay heat must 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 TSs 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. In all cases, except for the use of the most conservative method, a justification for lower decay heat input must be provided.

- *Operator action (actually inaction) time is 10 minutes.*

Generally, operator action for transient mitigation or termination can be credited 10 minutes after being alerted to the problem. 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 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 correlation's 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 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.

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

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

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 (see footnote 4 on previous page). 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 from the result of an energy addition transient analysis performed for a number of setpoints. 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 standards for spring loaded safety relief valves and/or manufacturer's recommendations, whichever is more conservative. Typically, 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. 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 standards for spring loaded safety relief valves and/or manufacturer's recommendations, whichever is more conservative. 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. Due to their dedicated use, 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.

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 case of the Pressurizer relief valve, the pressure obtained at the valve inlet needs to be adjusted to the Pressurizer by adding the inlet piping pressure drop.

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 analytical model for analysis of this event under water-solid conditions that CE uses includes 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.

The model assumes that the primary coolant "inside" the SDC nozzle(s) (this includes the RV and portions of the hot and cold legs), that has been circulated through the SDC system is initially at a lower (called 'primary') temperature. Similarly, the primary coolant "outside" of the SDC nozzles, which has not been circulated, is initially at a higher temperature, which is assumed to equal the SG secondary temperature. In addition to the primary volumes in the SGs, the "outside" primary volume includes that in the RCP suction legs, RCPs, and the portions of the hot and cold legs not accounted for in the "inside" volume.

A five-node system is used to model the RCS:

1) SG in the operating RCP loop;

2) SG in the non-operating RCP loop;

3) RV annulus region;

4) Reactor core; and

5) RV upper plenum.

Further details regarding this methodology are provided in Appendix D.

Subsequent to calculating the initial conditions and constants, a time dependent technique is used to model the primary coolant temperature throughout the RCS resulting from the RCP start. At the first time increment, average property values at each of the nodes are recalculated, using appropriate energy balances, to compensate for the displaced volume elements.

Heat transferred in the SGs is calculated. The SG heat transfer area includes the surface area of the active portions of the tubes, with no tubes plugged. The heat transfer rate is a function of the average bulk ΔT (secondary-to-primary temperature difference). The overall heat transfer coefficient for each SG is invariant with time and is based on the initial flow through the respective SG. All energy transferred from the secondary side is absorbed totally by the primary coolant, with the metal masses of the RCPB neglected as heat sinks.

The total system energy content is updated to include Pressurizer heaters, decay heat, and RCP heat, in addition to the heat input from the SGs. Finally, RCS pressure is computed as a function of total system energy content and specific volume. Each time increment's calculation concludes with a check on convergence of the primary and secondary temperatures in SG.

At the beginning of each time increment RCS pressure is compared to the relief valve set pressure. When the RCS is a closed system (no mass flux), the specific volume remains constant and the analysis proceeds as described above. If the set pressure has been exceeded, then the relief valve discharges mass from the system and the computed RCS pressure after each time increment accounts for the mass release and energy convection. Liquid relief capacities are dependent on system pressure and temperature.

The model calculates fluid temperatures, specific volumes, relief valve discharge flow rates (after the valve opens), and other transient parameters every time increment.

Computer codes or hand calculations can be utilized for analyses of this event under other initial conditions, provided that the initial conditions are controlled as LTOP limitations in the TS. If analysis methods change, they must be approved by the NRC prior to use.

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 basis for such an analysis is to prevent reaching the relief valve setpoint, this event (RCP start with a steam bubble) must not be considered among the design basis overpressure events. 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.

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. That flow rate will have to be discharged by the relief valve. 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.

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 thè 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 requires, 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

The LTOP methodology for CE NSSS designs does not consider the consequential effects of minimum transient pressure on RCP shaft seals because the seal design in use at CE NSSS designs is not susceptible to catastrophic failure due to operation at low system pressure[4]. The robust design and the operating experience of seals in use allow operation at low system pressure for a reasonable period of time without resulting in excessive coolant leakage or catastrophic failure. For example, the RCP seals would not be adversely affected by conservatively assuming no operator action to recognize or respond to the transient for one hour. It is anticipated that the RCP seals will operate at a considerably longer period of time at low pressure without excessive leakage or catastrophic failure.

The RCP seals in use in CE NSSS designs are from one of three manufacturers, but are 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. The RCP seal manufacturers specify minimum recommended operating pressure limits for the RCP seals based upon desired seal operating life. The recommended operating pressure limit is not based on failure limitations. The typical minimum operating pressure is 200 to 250 psig at the RCP suction. Extended operation of these RCP seals at a pressure below the recommended minimum limit may result in earlier seal replacement due to accelerated wear. Low pressure operation (even at zero pressure) will not promote a failure mechanism other than accelerated seal face wear. The accelerated wear on the RCP seal faces can occur when hydraulic pressure is less than the mechanical pressure exerted on the faces. For the duration of the LTOP transient, the RCP seals may experience some premature wear. But such operation will not result in a complete seal failure. Even in the

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

unlikely event a single stage were to fail, the remaining stages in the multiple stage design would prevent a loss of RCP seal pressure retention function and thereby prevent excessive leakage. This minimum operating pressure is factored into the RCP operating limits maintained in the plant operating procedures. 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.

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 TSs 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 TSs 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, a flow induced pressure drop between the RV inlet nozzle and the surge line nozzle in the hot leg. The 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 TSs to operate within a temperature range must be accounted for in determining the pressure drop.

For the existing TSs, 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 needs to be considered 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). Temperature instrumentation uncertainty is included in all cases where temperature related operating restrictions are applied. 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 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 (see Section 5.3.1.7), 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 by establishing a sufficient vent area.

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.

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.

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

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

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. $\Delta\, RT_{NDT}$ 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].

[5] When using the generic value for welds made using Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes, 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 presents the analytical techniques and methodology for developing beltline P-T limits that are utilized in the composite RCS operating limits. The method is directly applicable to heatup, cooldown and inservice hydrostatic tests.

5.1 GENERAL

The analytical procedure for developing operational P-T limits for the RV beltline utilizes the methods of Linear Elastic Fracture Mechanics (LEFM) found in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G (Reference 10), in accordance with the requirements of 10 CFR Part 50 Appendix G (Reference 1). For these analyses, the Mode I (opening mode) stress intensity factors are used for the solution basis.

The general method utilizes Linear Elastic Fracture Mechanics procedures, which relates the size of a flaw with the allowable loading that precludes crack initiation. This relation is based upon a mathematical stress analysis of the beltline material fracture toughness properties as prescribed in Appendix G to Section XI of the ASME Code.

The RV beltline 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 recognized. The above flaw geometry and orientation is the postulated defect size (reference flaw) described in Appendix G to Section XI of the ASME Code.

At each of the postulated flaw locations, the Mode I stress intensity factor, K_I, produced by each of the specified loadings is calculated and the summation of the K_I values is compared to a reference stress intensity, K_{IR}. K_{IR} is the critical value of K_I for the material and temperature involved. The result of this method is a relation of pressure versus temperature for each RV operating period that precludes brittle fracture. K_{IR} is obtained from a reference fracture toughness curve for low alloy reactor pressure vessel steels as defined in Appendix G to Section XI of the ASME Code. This governing curve is defined by the following expression:

$$K_{IR} = 26.78 + 1.223 \exp[0.0145(T\text{-}ART + 160)] \text{ ksi}\sqrt{\text{in}}$$

where,

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

T = temperature at the postulated crack tip, °F

ART = adjusted reference temperature at the postulated crack tip, °F (determined in accordance with Section 4.0)

For any instant during the postulated heatup or cooldown, K_{IR} is calculated at the metal temperature at the tip of the flaw, and the value of ART at that flaw location. Also, for any instant during the heatup or cooldown the temperature gradients across the RV wall are calculated (see Section 5.3) and the corresponding thermal stress intensity factor, K_{IT}, is determined. Through the use of superposition, the thermal stress intensity is subtracted from the available K_{IR} to determine the allowable pressure stress intensity factor and consequently the allowable pressure.

In accordance with the ASME Code Section XI Appendix G requirements, the general equations for determining the allowable pressure for any assumed rate of temperature change during Service Level A and B operation are:

$$2K_{IM} + K_{IT} < K_{IR} \quad (1)$$

$$1.5K_{IM} + K_{IT} < K_{IR} \text{ (Inservice Hydrostatic Test)}$$

where,

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

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

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

The general approach described above is the basis for P-T limit development for most CE NSSS designs.

5.2 DETERMINATION OF THE MAXIMUM STRESS INTENSITY VALUES

Practices, methodologies and techniques that are utilized in the development of the P-T limits, along with justification of the aforementioned, are described briefly herein. Detailed technical descriptions of the pertinent items are given in Sections 5.3 and 5.4. These limits have been developed to meet the requirements of 10 CFR 50 Appendix G.

A brief technical description of the procedures practiced by CE to develop brittle fracture limits for the CE NSSS design is given for the required components of the RCPB. These techniques are applicable to all CE NSSSs. These techniques have been applied to nuclear power plants designed to ASME Code editions later than the Summer 1972 Addenda since the incorporation of Appendix G to 10 CFR 50 in 1973. These analytical techniques are based partially on LEFM and provide 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

The vessel is the only location 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.

The RV regions considered in the analysis to establish brittle fracture limits are as follows:

1a. Beltline
1b. Vessel Wall Transition
1c. Bottom Head Juncture
1d. Core Stabilizer Lugs
1e. Flange Region
1f. Inlet Nozzle
1g. Outlet Nozzle

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. For some plant designs, the beltline region may also include the vessel wall transition. Typically, in either case, the material with the highest ART value falls within the cylindrical shell region below the vessel wall transition.

These locations have been analyzed utilizing the principles of LEFM described by Appendix G to Section XI of the ASME Code. These analyses considered plant heatup, plant cooldown and an isothermal leak test. A brief description of the general criteria follows.

5.2.1 GENERAL METHOD

In accordance with Appendix G, Section XI of the ASME Code (Reference 10), the Mode I (opening mode) stress intensity factor, K_I, is utilized and calculated at numerous intervals throughout the transient. The K_I is calculated at the crack tip of a postulated flaw. The postulated flaw size for the considered locations, except the flange and nozzles, are assumed to have a depth equal to one-fourth of the section thickness and a length equal to 1-1/2 times the depth. At each of these structural locations, flaws are analyzed on the inside surface for cooldown transient events and at both the inside and outside surfaces for heatup transient events.

The determination of the applied K_I is based on the results of a two dimensional heat transfer analysis and consideration of the primary membrane stress, σ_{pm}, primary bending stress, σ_{pb}, secondary membrane stress, σ_{sm}, and secondary bending stress, σ_{sb}. The resulting K_I for each component of stress can be calculated as follows:

$$K_{Im_i} = M_m \text{ x (membrane stress) } \sigma_{im} \text{ where } i = p \text{ or } s$$
$$K_{Ib_i} = M_b \text{ x (bending stress) } \sigma_{ib} \text{ where } i = p \text{ or } s$$

where M_m and M_b are defined in Appendix G, Section XI of the ASME Code (Reference 10). For computational simplicity, equations A3-4 and A3-6 of WRC Bulletin 175 (Reference 14) were utilized, where M_t and M_b is equivalent to M_m and M_b in Appendix G of the ASME Code (Reference 10), as follows:

$$M_t = \frac{1.1M_K\sqrt{\pi}}{2\sqrt{Q}}\sqrt{T} \quad \text{and}$$

$$M_b = \left(\frac{M_B\sqrt{\pi}}{2\sqrt{Q}}\right)\sqrt{T}$$

where:

M_K, M_B = correction factors dependent on the ratios of crack depth to section thickness and crack depth to crack length (Figures A3-1, A3-2, Reference 14)
Q = the flaw shape factor modified for plastic zone size
T = the section thickness (in).

$$K_{Im_i} = M_t \times \sigma_{im} = \frac{1.1M_K\sqrt{\pi}}{2\sqrt{Q}}\sqrt{T}\,\sigma_{im} \qquad \text{where } i = p \text{ or } s$$

$$K_{Ib_i} = M_b \times \sigma_{ib} = \frac{M_B\sqrt{\pi}}{2\sqrt{Q}}\sqrt{T}\,\sigma_{ib} \qquad \text{where } i = p \text{ or } s$$

For each point in the transient analyzed, the allowable pressure is determined by comparing the reference stress intensity, K_{IR}, to the applied stress intensity with a conservative factor of safety. The value of K_{IR} is obtained at the crack tip location based on the crack tip temperature for the specific time point in the transient and determined based on the following equation:

$$K_{IR} = 26.78 + 1.223 \exp[0.0145(T-RT_{NDT} + 160)] \text{ ksi}\sqrt{\text{in}}$$

where,

T = crack tip temperature (°F) at 1/4T and 3/4T locations
RT_{NDT} = reference nil ductility temperature at each cracktip location

For plant heatup and plant cooldown, the following expression is used to determine the allowable pressure:

$$K_{IR} > 2.0\, K_{Ip} + K_{IT}$$

Substituting,

$$K_{IR} > 2.0\, \underbrace{\{K_{Im_P} + K_{Ib_P}\}}_{\text{PRIMARY}} + \underbrace{\{K_{Im_S} + K_{Ib_S}\}}_{\text{SECONDARY}}$$

For leak tests, the expression utilized to calculate the K_I due to test pressure is:

$$K_{IR} > 1.5\, K_{Ip} + K_{IT}$$

Substituting,

$$K_{IR} > 1.5\ \{K_{Im_P} + K_{Ib_P}\} + \{K_{Im_S} + K_{Ib_S}\}$$

PRIMARY SECONDARY

Table 1 of 10 CFR Part 50, Appendix G outlines the pressure and temperature requirements for the reactor pressure vessel for the normal and hydrostatic pressure and leak tests operating conditions. The table 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 of 10 CFR Part 50, Appendix G, 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.

The minimum temperature requirements for the RV, 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 ART 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 ART for the limiting material in the closure flange region plus 90 °F.

- For normal operations, when the RCS pressure is less 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 ART 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 ART 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 ART 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 ART 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.

5.2.2 FLANGES

The flange is analyzed assuming a flaw size of 0.75 inch and is smaller than a one-quarter depth flaw. This smaller flaw size is permitted by Article G-2120 of Appendix G to Section XI of the ASME Code and is based on the ability to confidently detect this flaw size utilizing in-shop non-destructive examination (NDE) techniques (e.g., radiography, ultrasonic testing, etc.) and is consistent with the acceptance standards of Sub-Article NB-5320 of Section III to the ASME Code.

The applied K_I is determined utilizing equations A3-1 and A3-2 along with Figures A3-1 and A3-2 from NRC approved WRC Bulletin 175 (Reference 14). The remainder of the procedures, as described previously are also applicable to the flange region.

5.2.3 NOZZLES

In the case of the primary inlet and outlet nozzles the method described in Appendix 5 to NRC approved WRC Bulletin 175 (Reference 14), K_I Calculation Method for Nozzle, was utilized.

In this analysis the postulated flaw size was equal to one-tenth of the vessel wall thickness and located on the inside corner of the nozzle adjacent the vessel wall. Again, the flaw size is confidently detectable with the in-shop NDE techniques and consistent with the acceptance standards of Sub-Article NB-5320 of Section III to the ASME Code.

The applied K_I due to membrane stress are determined utilizing Equation A5-1 in conjunction with Figure A5-1 (both are from Reference 14). The bending stress intensity factor is calculated in the same manner as the other locations.

The solution for the allowable pressure is still based on K_{IR} as the maximum allowable stress intensity factor for the particular crack tip temperature. The relations previously cited for heatup and cooldown, and leak test were applied in determining the applicable limits.

The results of these analyses, in the unirradiated condition, show that for heatup, cooldown and isothermal leak test, the limiting locations are the vessel shell at the vessel flange, the inlet nozzle and the upper shell at the vessel wall transition, respectively.

5.2.4 BELTLINE

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 material. Therefore, the beltline region will likely become the controlling location when compared to the other RCS locations analyzed. CE considers the beltline region to be controlling, that is, the most limiting with respect to allowable pressure at any specific temperature, when the shift in RT_{NDT} due to neutron radiation in the beltline causes the ART to be greater than the unirradiated RT_{NDT} of the surrounding locations. 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 based on procedures described in Sections 5.3 and 5.4 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.

The operational limits as indicated in the control room account for the temperature differential between the RV base metal and the reactor coolant bulk fluid temperature. Corrections for elevation and flow induced pressure differences between the RV beltline and Pressurizer are included. Pressurizer pressure indicator loop uncertainties are also included and consequently, the limits are provided on coordinates of indicated Pressurizer pressure versus indicated RCS (cold leg) temperature.

5.3 PRESSURE-TEMPERATURE LIMIT GENERATION METHODS

5.3.1 GENERAL DESCRIPTION OF P-T LIMITS GENERATION

5.3.1.1 Process Description

PT limits are generated via the following approach 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 Appendix G, ASME Section XI requirement 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 stress analysis of the RV beltline 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).

The general process to generate PT Limits is as follows:

a) Determine the limiting ART for the postulated 1/4T and 3/4T crack tip locations of the RV.

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

d) Calculate 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 which does not exceed this requirement.

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 which does not exceed this requirement *(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.

5.3.1.2 Regulatory Requirement

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 Service Level A and B *(Normal and Upset Loads, respectively)* operation is:

$2K_{IM} + K_{IT} < K_{IR}$ (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 $\sqrt{in}$

5.3.1.3 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} that is defined as 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:

$$K_{IA} = 26.78 + 1.223e^{[0.0145(T-RT_{NDT} + 160)]} \qquad \text{Reference 10}$$

$$K_{IC} = 33.20 + 2.806e^{[0.0200(T-RT_{NDT} + 100)]} \qquad \text{Reference 10}$$

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 at 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 adjusted RT_{NDT} at that flaw location.

Note: 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 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.

5.3.1.4 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:

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

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 derived from the calculated allowable pressure stress intensity factor, K_{IM}, shown above. The value of K_{IM} will depend on the approaches discussed in Sections 5.3.3 through 5.3.5.

5.3.1.5 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.

5.3.1.6 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.

5.3.1.7 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. However, uncertainties for instrumentation error, elevation, and flow induced differential pressure corrections are not accounted for and must be included by the plant when final P-T limits are developed.

5.3.2 THERMAL ANALYSIS METHODOLOGY

The first step in P-T limits generation 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,

$$\rho C \frac{\partial T}{\partial t} = K\left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right] \qquad \text{(Reference 19)}$$

subject to the following boundary conditions at the inside and outside wall surface locations (Reference 19, p. 109):

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

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

where,

ρ	=	density, lb/ft^3
C	=	specific heat, Btu/lb-°F
K	=	thermal conductivity, Btu/hr-ft-°F
T	=	vessel wall temperature, °F
r	=	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 wall, 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.

5.3.3 CE NSSS P-T CURVE METHOD

For 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). This methodology generates results at the crack tips based on unit loads of pressure and temperature as described in the following sections.

5.3.3.1 Calculation of Thermal Stress Intensity Factors, K_{IT}

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). The alternate method employed uses a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients are calculated using a 2-dimensional finite element model of the RV.

The superposition technique employed is temperature profile based rather than the stress profile based which is typically used. A third order polynomial fit to the temperature distributions in the wall was used and is given by:

$$T(x) = C_o + C_1\,(1-{}^{x}/_{h}) + C_2\,(1-{}^{x}/_{h})^2 + C_3\,(1-{}^{x}/_{h})^3$$

where,			
	$T(x)$	=	Temperature at radial location x from inside wall surface
	C_0, C_1, C_2, C_3	=	Coefficients in polynomial fit
	x	=	Distance through beltline wall, in
	h	=	Beltline wall thickness, in

These polynomial fit coefficients are utilized in determination of the applied stress intensity.

In the following section, temperature based influence coefficients, K_I^*, for determination of the thermal stress intensity factor, K_{IT}, are discussed. The influence coefficients are dependent upon the geometrical parameters associated with the maximum postulated defect, and the geometry of the RV beltline region (i.e., r_o/r_i, a/c, a/t), along with the unit loading.

5.3.3.2 Calculation of Allowable Pressure

As presented above, the Appendix G equation relating K_{IM}, K_{IT}, and K_{IR} is rearranged 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:

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

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$\sqrt{in}$

For pressure loadings, unit values of the load distributions were used to compute the influence coefficients. The unit value chosen for internal pressure was 1000 psi.

The general equation to compute the Mode I stress intensity factors for thermal and pressure loading conditions is as follows:

$$K_I(a) = \sum_{i=0}^{3} C_i K_i^* \sqrt{\pi a}$$

where,

$K_I(a)$ = Total applied stress intensity factor due to loading condition at crack depth, a

C_i = Polynomial coefficients from the curve fit to the temperature or stress distribution through the vessel wall

K_i^* = Fracture mechanics influence coefficients for a specified loading condition for each term of the polynomial expression for the temperature or stress distribution through the vessel wall

a = crack depth, in

The K_I for each loading condition is then summed and compared to the allowable K_{IR} to determine the allowable pressure.

The allowable pressure is derived from the calculated allowable pressure stress intensity factor, K_{IM}, shown above. For calculation purposes, the allowable pressure can be represented by the following expression once the allowable pressure stress intensity factor is determined.

$$\text{P-Allowable} = \frac{K_{IM}}{K_{IM}^*}$$

where,

P-Allowable = allowable pressure as a function of time or coolant temperature, Ksi

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

K_{IM}^* = pressure stress intensity factor for 1000 psia internal pressure as determined from a finite element model, Ksi $\sqrt{in}$

5.3.4 *STANDARD ASME P-T CURVE METHOD*

As intended, ASME Section XI, Appendix G (Reference 10), provides sufficient guidance and direction through figures and text to perform P-T calculations in a straight-forward fashion. The following outlines the ASME Appendix G calculational procedure used in this report to generate the allowable pressure. Beginning with Equation (1) of G-2215, the general equation for determining the allowable pressure for any assumed rate of temperature change during Service Level A and B operation is:

$$2K_{IM} + K_{IT} < K_{IR}$$

then, solving for $K_{IM,}$ we have

$$K_{IM} < (K_{IR} - K_{IT}) / 2$$

where $K_{IM} = M_m * \sigma_m = M_m * Pr/t$

where $\sigma_m = Pr/t$ (Membrane hoop stress)

substituting and solving for P-Allowable (ksi), we have

$$\text{P-Allowable} < (K_{IR} - K_{IT})*t / (2*r*M_m)$$

where,

P-Allowable = Allowable pressure, Ksi

from Figure G-2214-1 1996 ASME Code,

$K_{IT} = M_t * \Delta T_w$

where M_t is obtained from figure as function of wall thickness at 1/4T depth, and

$\Delta T_w = T(OD) - T(ID)$ from Heat Transfer Analysis at each time point (Section 5.3.3.2)

K_{IR} = Reference stress intensity factor, Ksi $\sqrt{\text{in}}$, per Figure G-2210-1 *(for $K_{IR} = K_{IA}$)*

M_m = From formulas in G-2214.1 1996 ASME Code

t = Base Metal Wall Thickness, in

r = Base Metal Inner Radius, in

This formulation is used in conjunction with the basic data identified above, along with a common through-wall temperature analysis of the heatup and cooldown transients to generate P-Allowable.

5.4 TYPICAL PRESSURE-TEMPERATURE LIMITS

This section presents example P-T limits for the RV beltline region and the reactor flange region. These limits were developed using the methods described in Section 5.1 through 5.3 in conjunction with the following information.

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

Reactor Vessel Data

Design Pressure = [2500] psia
Operating Pressure = [2250] psia
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	Adjusted RT_{NDT} (°F)
1/4 T	[191.0]
3/4 T	[137.0]

Initial RT_{NDT}

Flange Region = [+80]°F
Piping, Pumps and Valves = [+90]° F

Pressure and Temperature Correction Factors

ΔT = [+6]° F
[For T_C < 200°F; ΔP = -77 psi (2 RCP's operating)]
[For $T_C \geq$ 200°F; ΔP = -69 psi (3 RCP's operating)]

5.4.1 BELTLINE LIMIT CURVES

The beltline P-T limits calculated for heatup and cooldown are depicted in Figures 5.1 through 5.4 and have been developed utilizing the CE NSSS methodology described in Section 5.1 through 5.3. 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. Also, these figures have been corrected to indicated Pressurizer pressure and cold leg temperature (T_C).

Depicted in Figure 5.5 is the beltline P-T curve for inservice hydrostatic test. This limit curve is typically developed for an isothermal condition. Again, this figure has been corrected to indicated Pressurizer pressure and cold leg temperature. The purpose of this figure is to establish the minimum temperature corresponding to the required hydrostatic test pressure. Note that CE's practice for CE NSSS designs is to recommend a minimum temperature for inservice hydrostatic test based on a test pressure corresponding to 1.1 times the design pressure.

5.4.2 *FLANGE LIMIT CURVES*

The vessel flange limits, resulting from the detailed analysis described in Section 5.2.2, are shown in Figure 5.6. This figure has been corrected to indicated Pressurizer pressure and cold leg temperature.

5.4.3 *COMPOSITE LIMIT CURVES*

The beltline P-T limits and flange P-T limits discussed in previous sections form the basis for the composite limit curves. In addition, the nozzle requirements described in Section 5.2.3 are also considered when developing the composite RCS P-T limits.

During the development of the composite limits, the heatup and cooldown rates are chosen based on numerous considerations. The issues involved in establishing these maximum rates include the impact on the operating window, the selection of the LTOP setpoint(s), the plant's physical limitations, and the economical impact associated with loss of electrical power generation. The relative importance of these items is different for each utility and therefore is not addressed directly in this document.

For the purpose of illustration, composite limits were developed for heatup and cooldown and are presented in Figures 5.7 and 5.8, respectively. These figures show arbitrary rates selected for heatup and cooldown that will be used to develop the PTLR figures. Included in the figures are all of the analyzed locations and additional requirements necessary to determine which specific location is controlling with respect to operating temperature.

Again, for the purpose of illustration, the minimum boltup temperature was conservatively established to be [80]°F and the lowest service temperature was established to be [196]°F. Both requirements are depicted as part of the composite heatup and cooldown limits.

The composite limit curve for inservice hydrostatic test is shown in Figure 5.9. The minimum temperature for inservice hydrostatic pressure test, [322]°F was established based on a test pressure of [2427] psia (1.1 times normal operating pressure).

The limitations associated with core critical operation are developed along with the PTLR figures. These are presented in Section 5.4.4.

5.4.4 OPERATIONAL LIMIT CURVES

The operational limits developed for utilities are based on the composite limits presented in the previous section. Typical representations of figures developed for inclusion in the PTLR are presented in Figures 5.10 and 5.11.

Figure 5.10 presents typical heatup limits developed to protect the RCS from brittle fracture. Included with the actual heatup limits are the limits representing inservice hydrostatic test and limits pertaining to core critical operation. The core critical limits were established based on the requirements given in Section 6.1. In addition, the allowable rates utilized in development of the heatup limits are also given as maximum heatup rates for the appropriate temperature range.

Figure 5.11 presents typical cooldown limits established to protect the RCS from brittle fracture. Again, limits representing inservice hydrostatic test are also present with the composite cooldown limits. The allowable rates, utilized to develop the cooldown limit curve, are also listed as maximum cooldown rates for the appropriate temperature range. The limitations for critical operation of the core are usually not presented as part of the cooldown PTLR figure.

5.4.5 SUMMARY

This section describes methodologies and practices utilized in the development of RCS P-T limits. The methodology was developed to meet the specific criteria of 10 CFR 50, Appendix G, Fracture Toughness Requirements and 10 CFR 50, Appendix A, Design Criterion 14 and Design Criterion 31.

The current requirements imposed by 10 CFR 50, Appendix G, apply to pressure-retaining components of the RCPB which are fabricated from ferritic material and apply to any condition of normal operation, including anticipated operational occurrences and system hydrostatic pressure tests. Section A.4 provides a list and an operational description of the conditions that require P-T limits.

The method and analytical procedures used in the development of the RCS P-T limits are based on linear elastic fracture mechanics techniques described in ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, Fracture Toughness Criteria for Protection Against Failure. As noted previously, the required loading conditions are described in Section A.4. As discussed in Section 5.2, the only component specifically requiring a LEFM analysis is the RV. Additional details on the RV locations that were analyzed and the technical methodology are also provided.

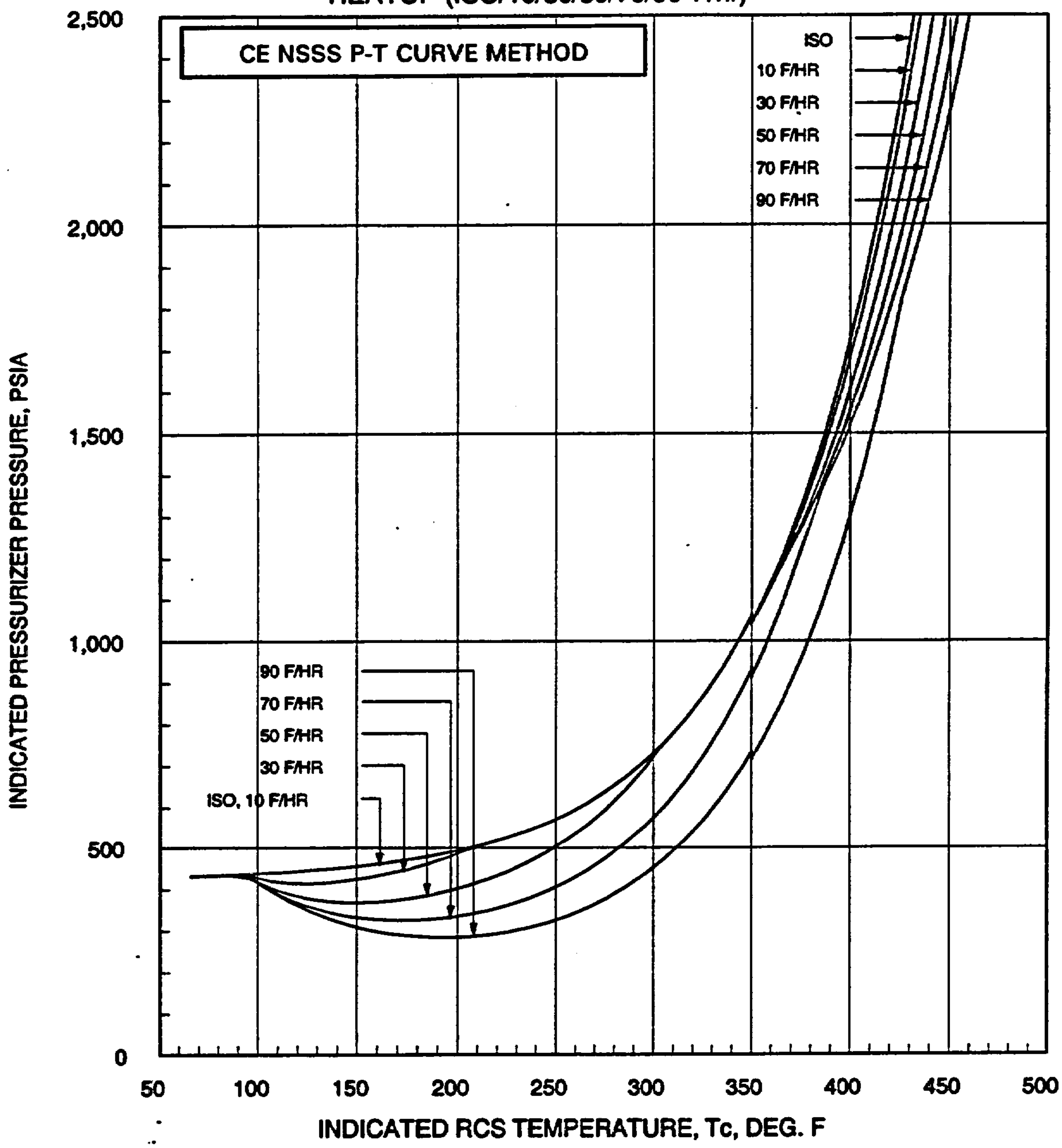
The results of the LEFM analysis performed for the RV provided the limiting locations in the unirradiated condition for heatup, cooldown and isothermal leak test. The limiting locations considered are the vessel shell at the vessel flange, the inlet nozzle and the vessel wall transition region. These results are considered in the development of composite RCS operating limits. Typically, when the RCS operating limits are developed for a specific time period, the beltline becomes the most limiting location in the RV because of the effects of neutron irradiation. Therefore, when RCS operating limits are developed, the beltline is analyzed considering the effect of neutron irradiation in accordance with Regulatory Guide 1.99 Revision 2 (see Section 4.0 and 7.0), and the vessel flange region is considered, as a minimum, per the requirements of 10 CFR 50 Appendix G (see Section 6.0).

To illustrate the application of these methodologies and practices, RCS P-T limits are discussed in Sections 5.1 through 5.4 for a typical plant. Included is a description of the process utilized to develop composite limits which protect the RCPB from brittle fracture and typical technical specification figures which specifically address the requirements of 10 CFR 50 Appendix G providing limits for normal operation, inservice hydrostatic test, and core critical operation.

FIGURE 5.1

APPENDIX G P-T LIMITS

HEATUP (ISO/10/30/50/70/90°F/hr)



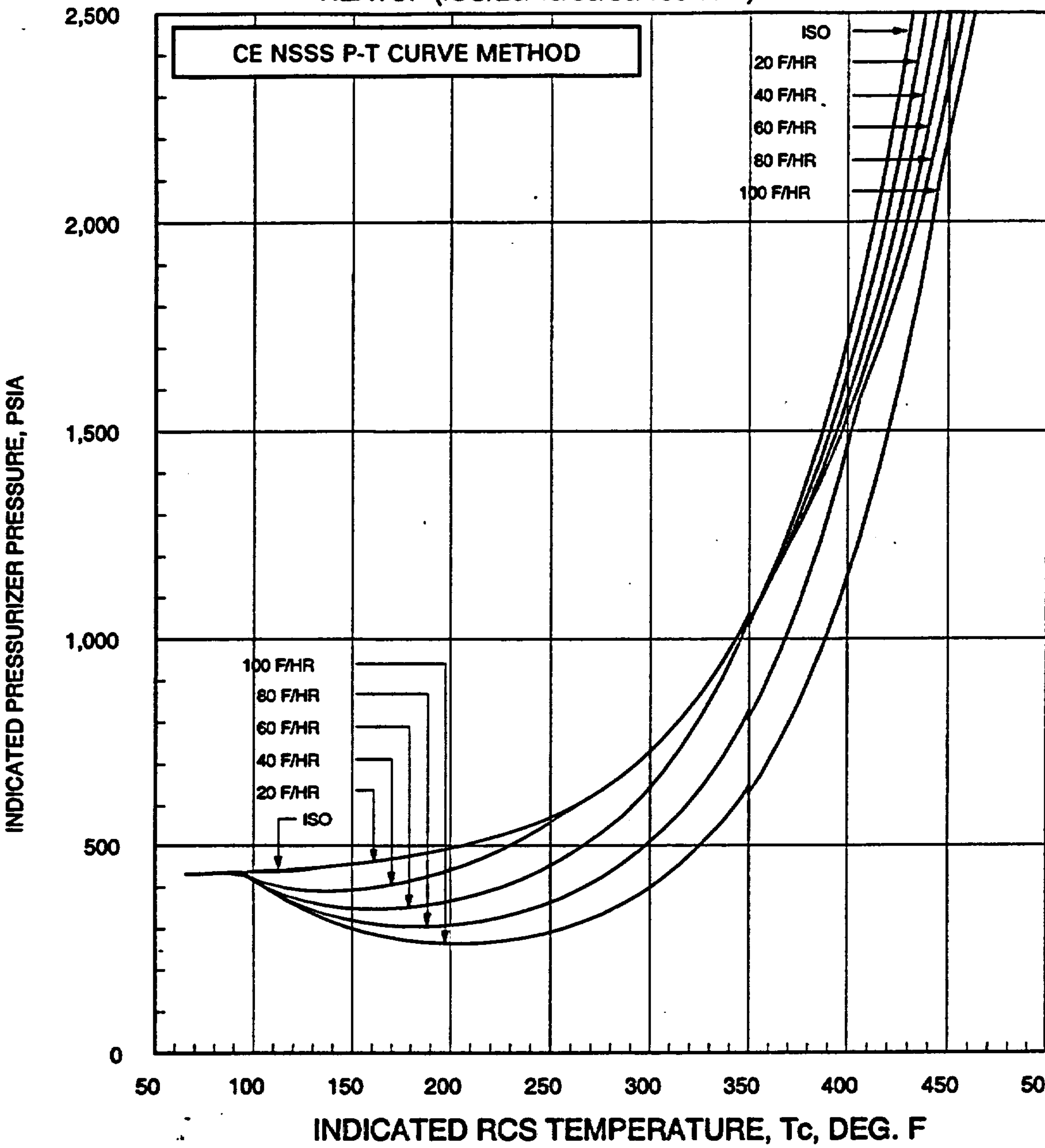
Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.2

APPENDIX G P-T LIMITS

HEATUP (ISO/20/40/60/80/100°F/hr)



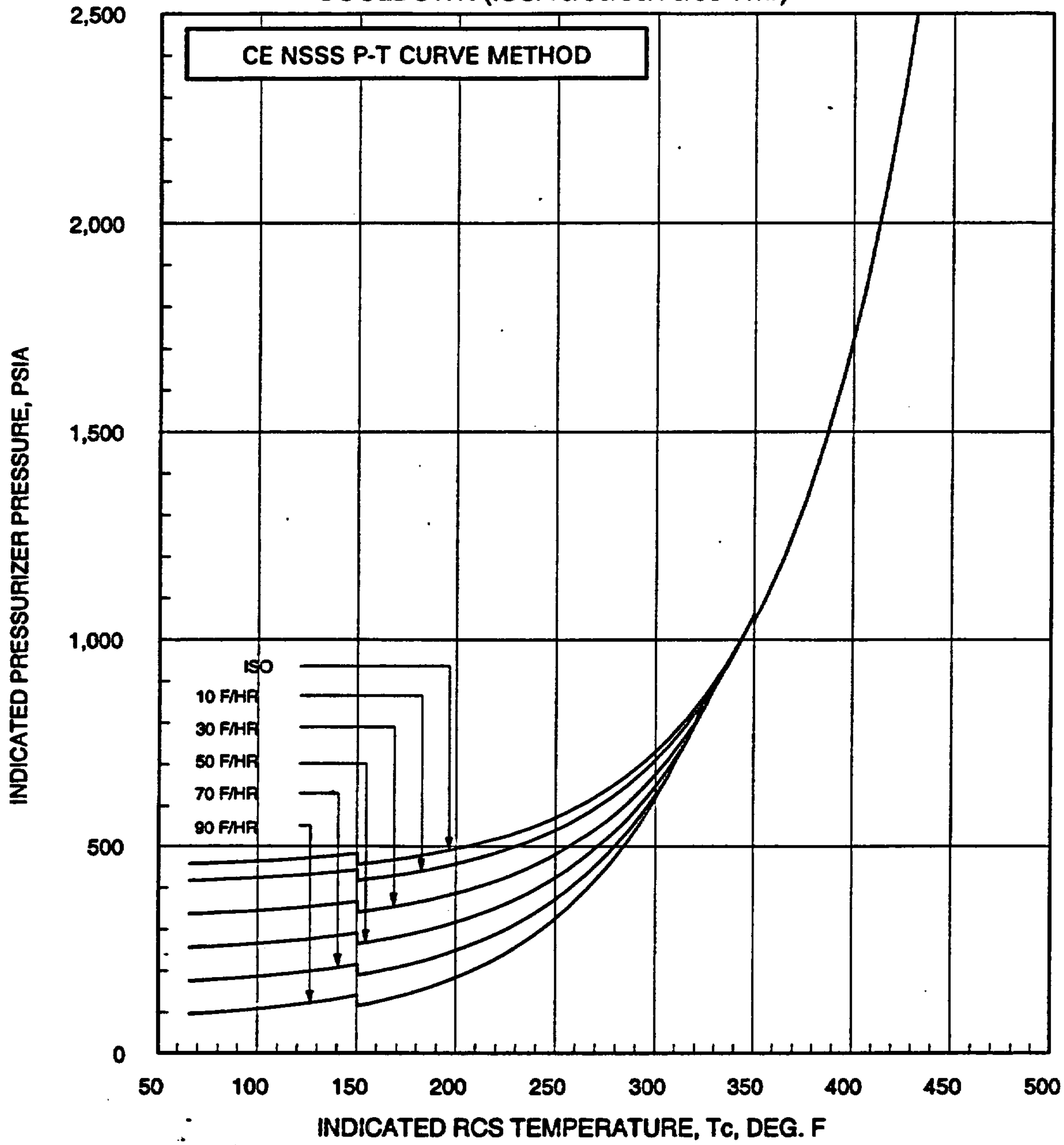
Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.3

APPENDIX G P-T LIMITS

COOLDOWN (ISO/10/30/50/70/90°F/hr)

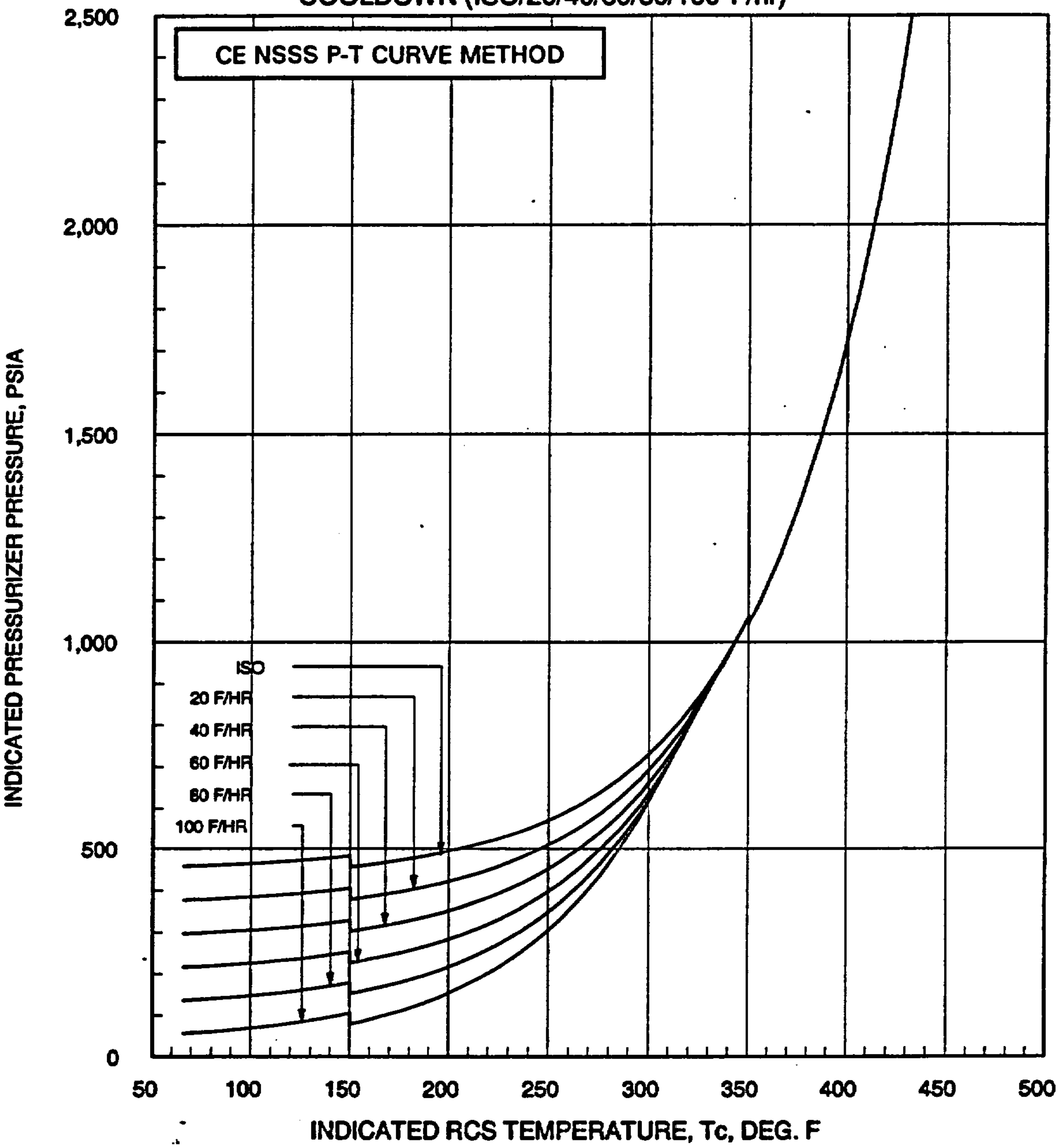


Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.4

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

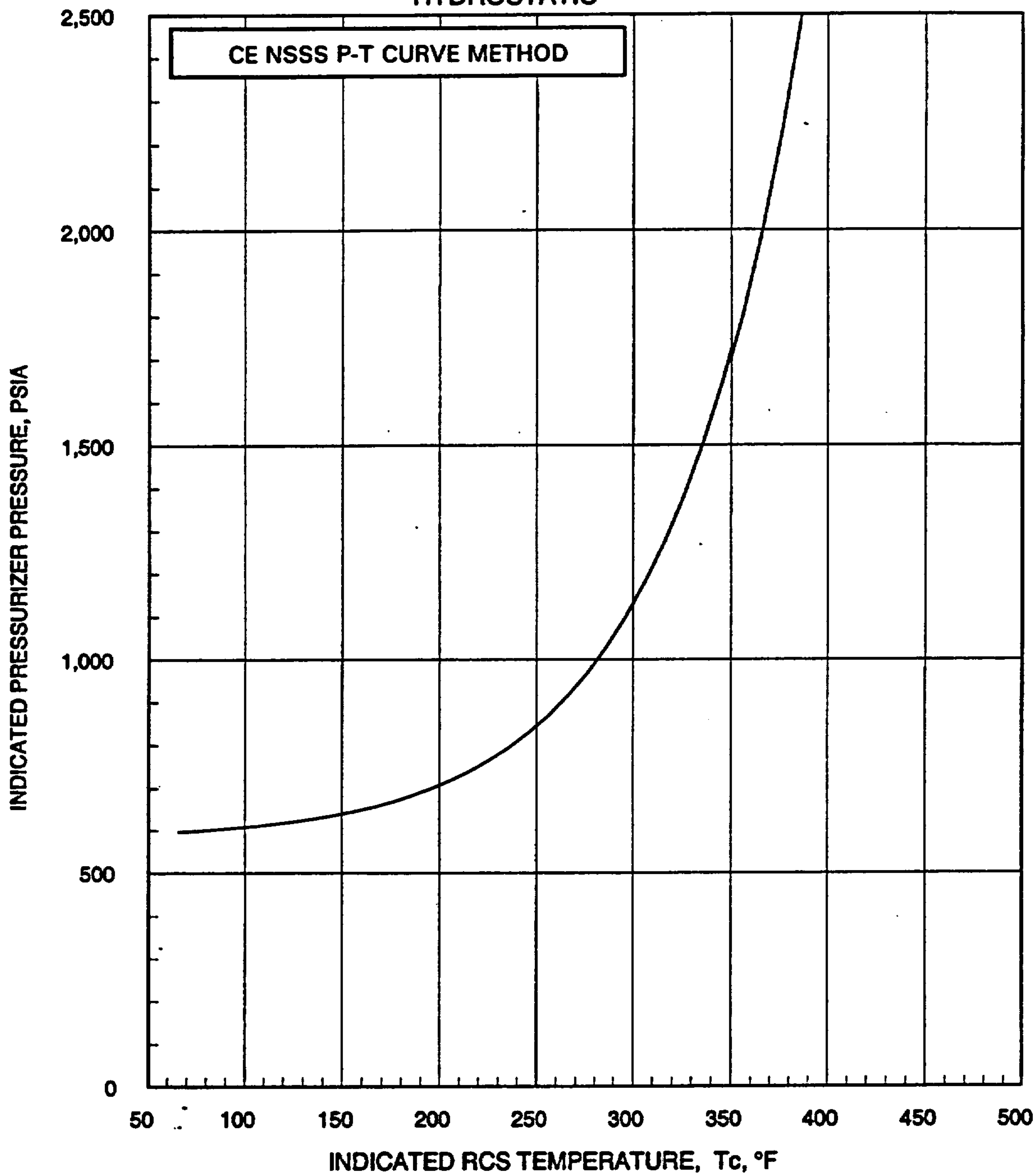


Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.5

APPENDIX G BELTLINE P-T LIMITS
HYDROSTATIC



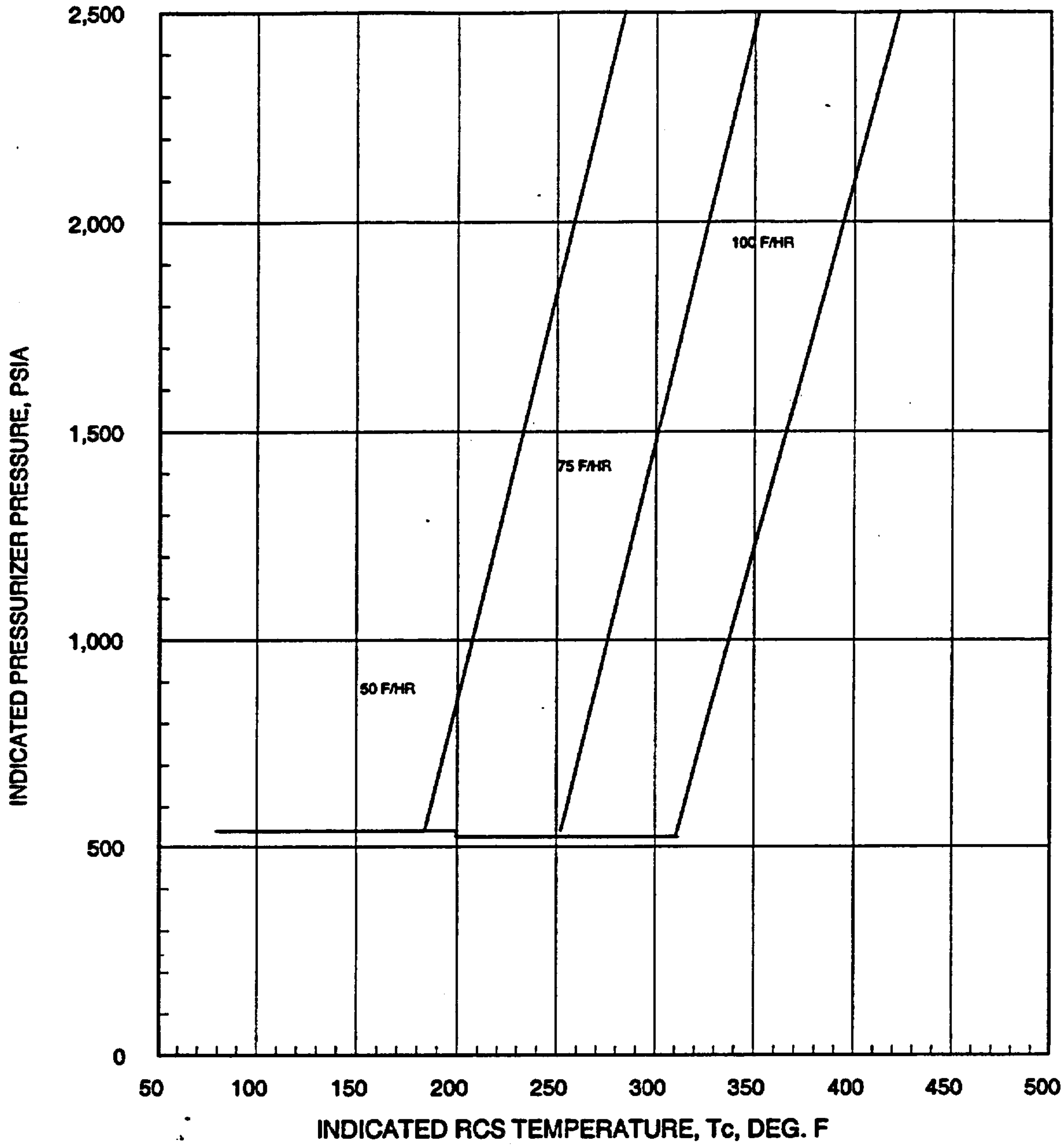
Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.6

APPENDIX G FLANGE LIMITS

HEATUP

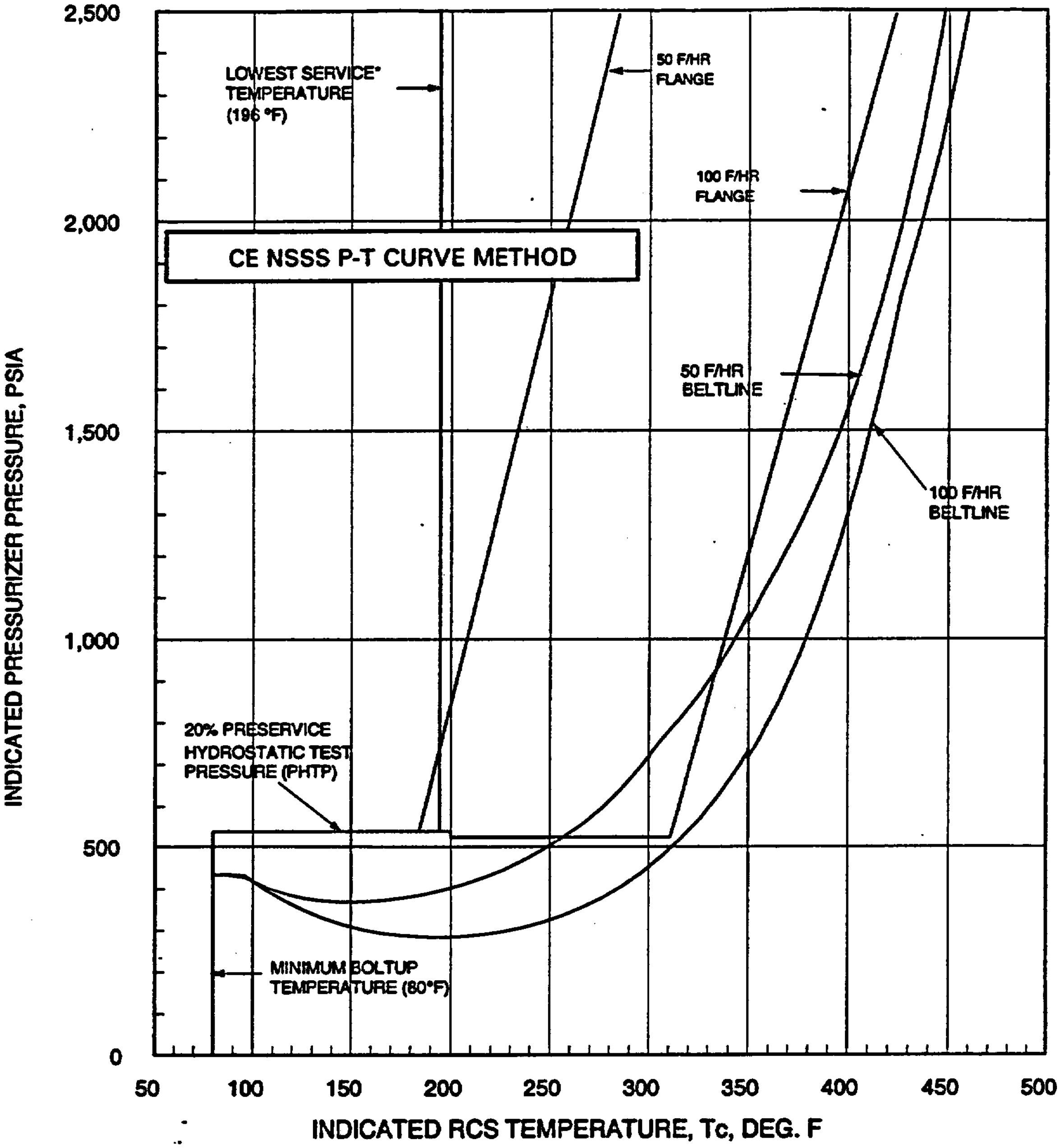


Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.7

COMPOSITE APPENDIX G P-T LIMITS
HEATUP



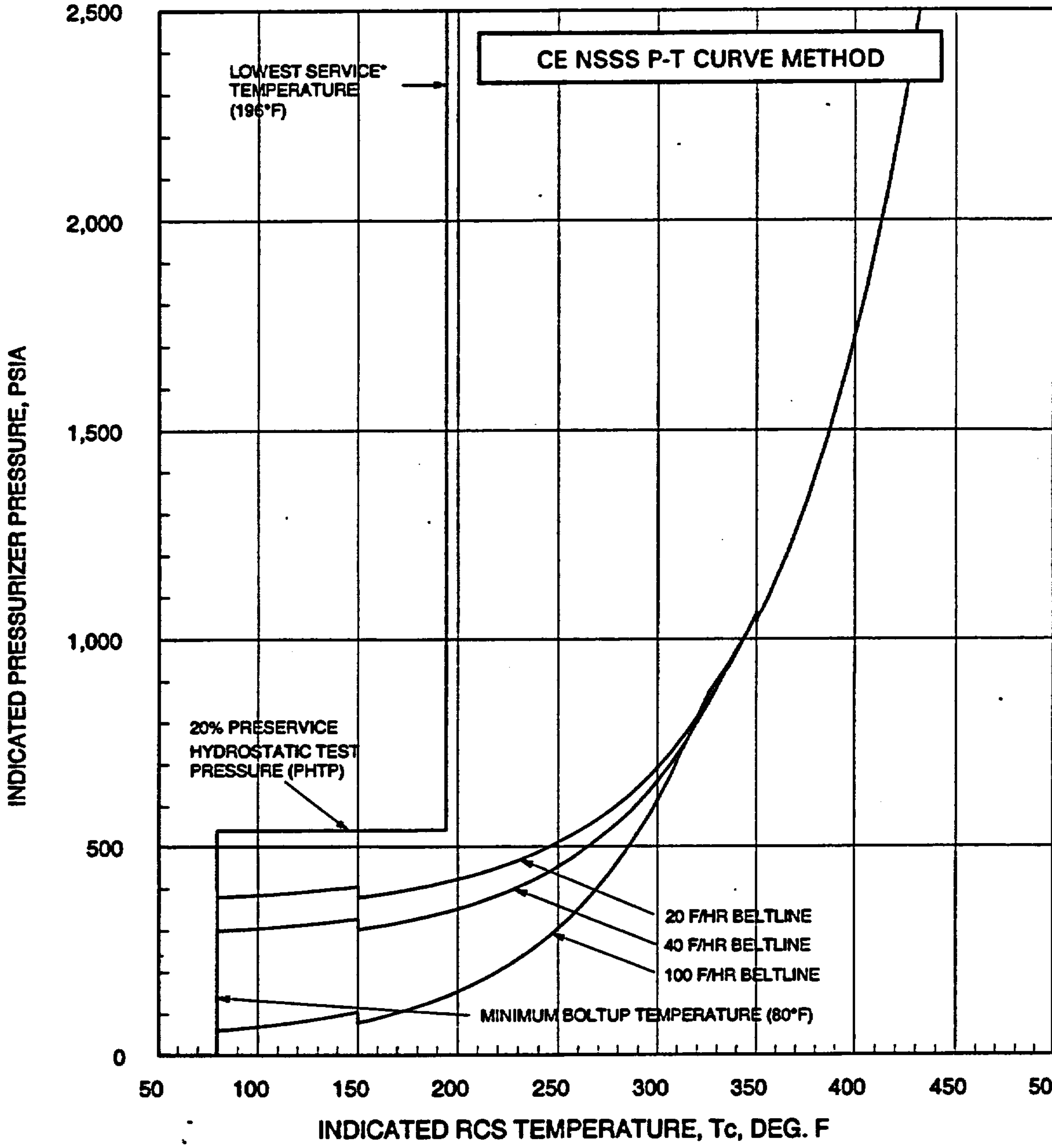
*This line is defined as the more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.8

COMPOSITE APPENDIX G P-T LIMITS
COOLDOWN



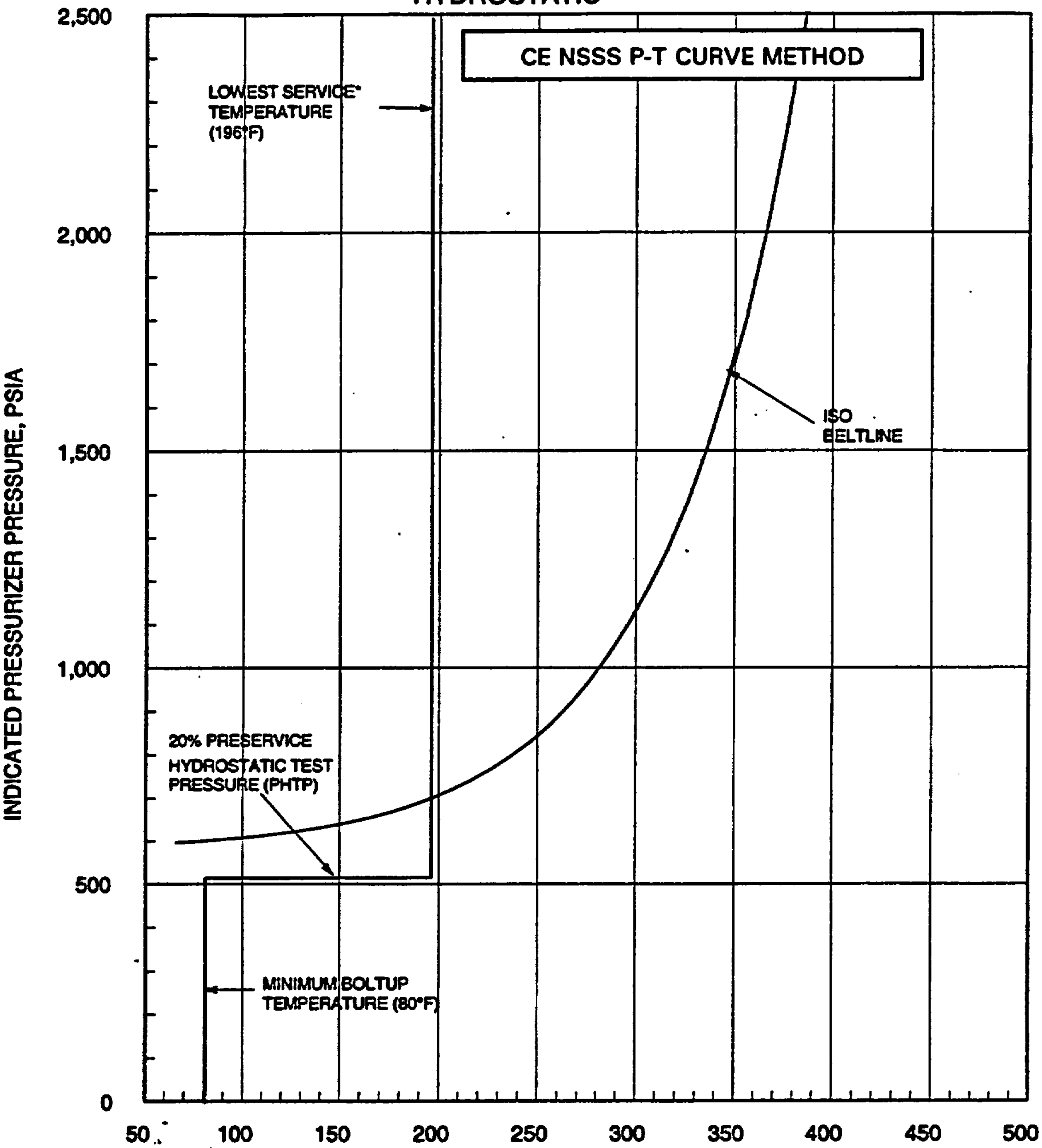
*This line is defined as the more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

Tc < 200°F, ΔP = -77 psi	
Tc ≥ 200°F, ΔP = -69 psi	
ΔT = +6°F	

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.9

COMPOSITE APPENDIX G P-T LIMITS
HYDROSTATIC



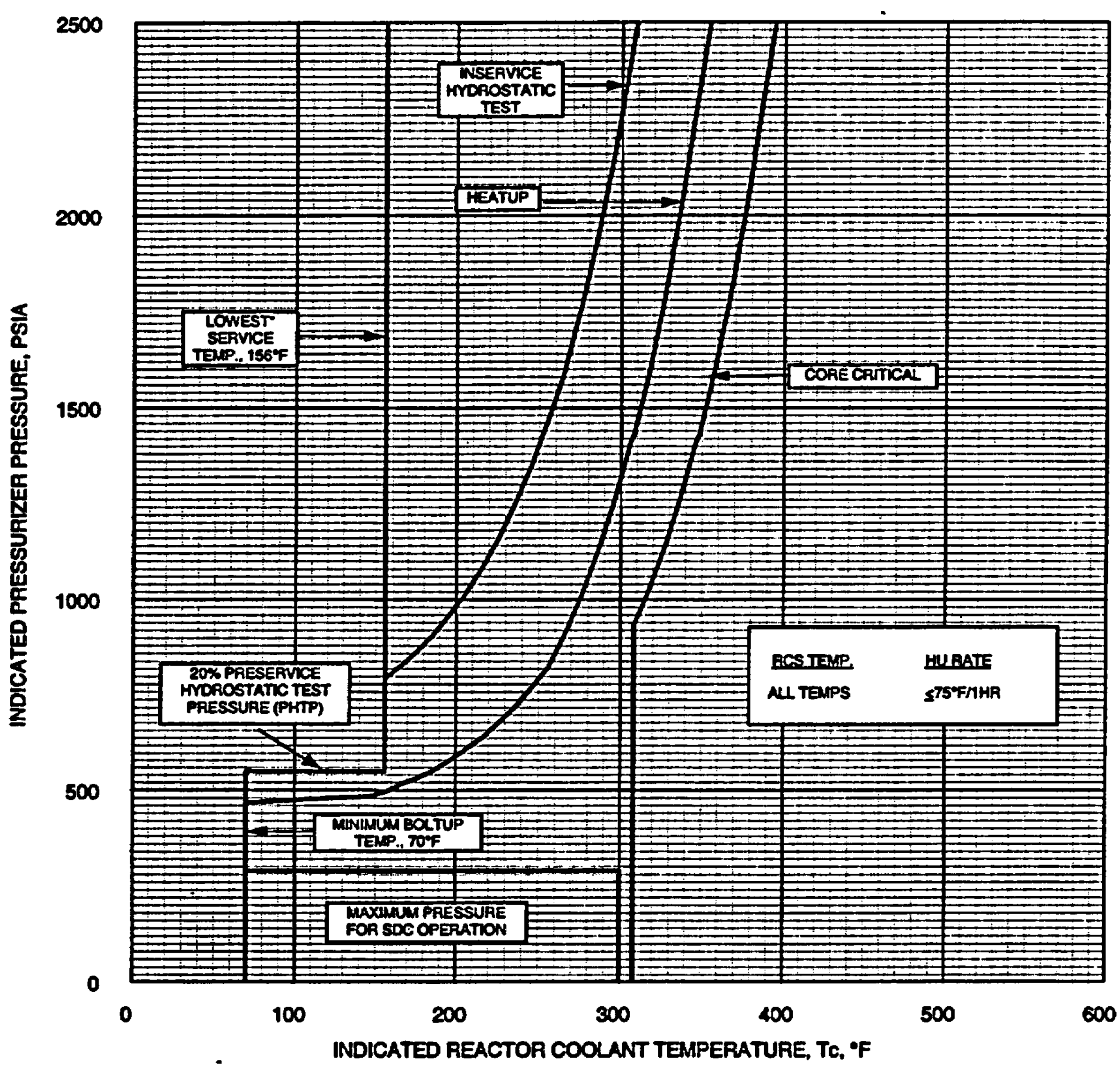
*This line is defined as the more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

Tc < 200°F, ΔP = -77 psi
Tc ≥ 200°F, ΔP = -69 psi
ΔT = +6°F

ART
1/4t = 191.0°F
3/4t = 137.0°F

FIGURE 5.10

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



*This line is defined as 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.

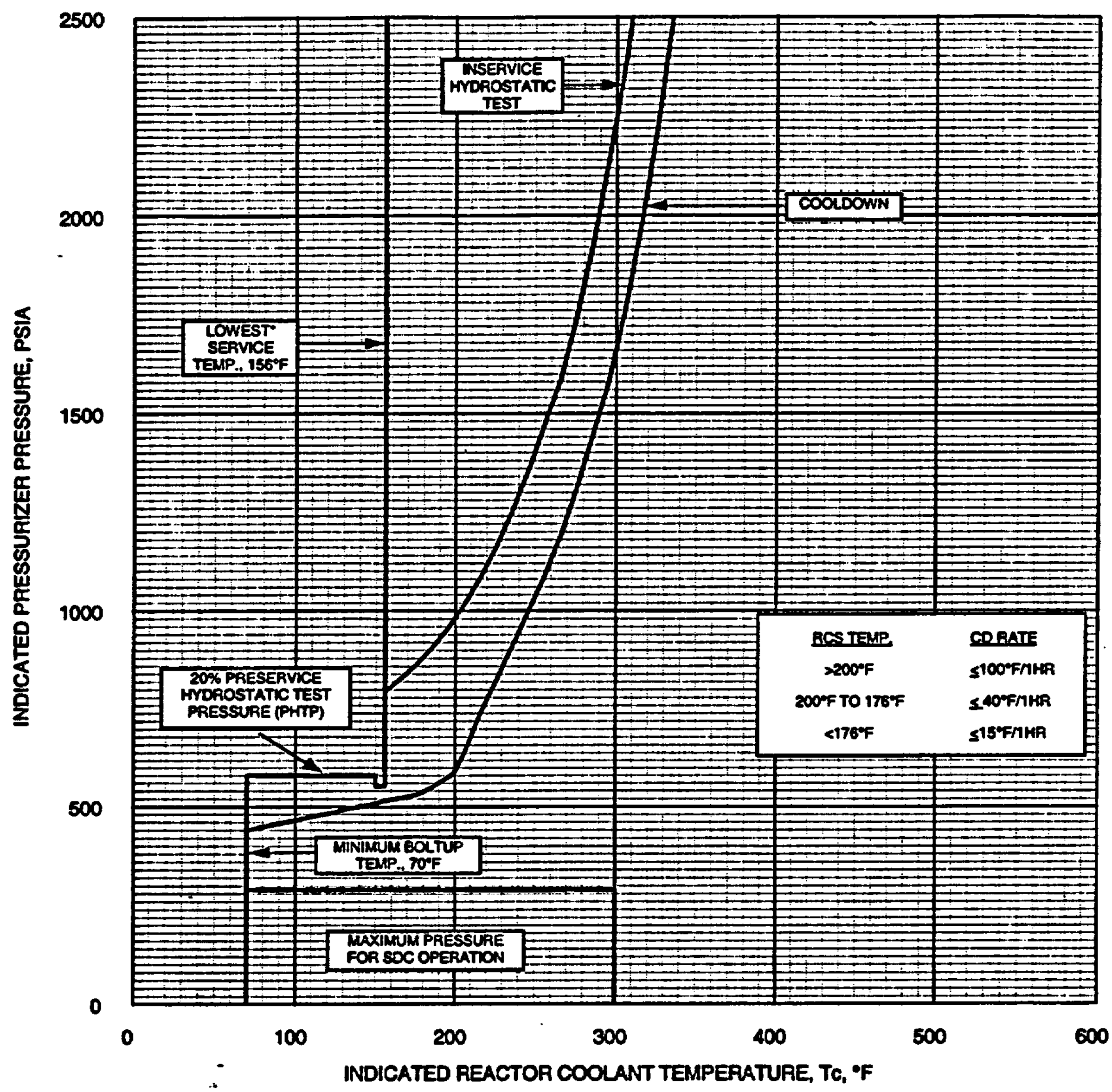
FIGURE 5.11

TYPICAL REACTOR COOLANT SYSTEM

PRESSURE TEMPERATURE LIMITS FOR

TECHNICAL SPECIFICATIONS

COOLDOWN



*This line is defined as 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.

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

6.1 INSERVICE HYDROSTATIC PRESSURE TEST AND CORE CRITICAL LIMITS

Both 10 CFR Part 50 Appendix G and the ASME Code, Section XI, Appendix G require the development of P-T limits which are applicable to inservice hydrostatic tests. For hydrostatic tests performed subsequent to loading fuel into the RV, prior to core criticality, the minimum test temperature is determined by evaluating K_I, the mode I stress intensity factors. The evaluation of K_I is performed in the same manner as that for normal operation heatup and cooldown conditions except the factor of safety applied to the pressure stress intensity factor is 1.5 versus 2.0. From this evaluation, a P-T limit that is applicable to inservice hydrostatic tests is established. The minimum temperature for the inservice hydrostatic test pressure can be established conservatively by determining that the test pressure corresponding to 1.1 times normal operating pressure and locating the corresponding temperature. Hydrostatic testing of the RV after achieving core criticality is not allowed.

The minimum temperature requirements for the RV, 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 <u>is not</u> critical, the minimum temperature requirement for the RV must be at least as high as the ART 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 <u>is not</u> critical, the minimum temperature requirement for the RV must be at least as high as the ART for the limiting material in the closure flange region plus 90 °F.

- For normal operations, when the RCS pressure is less or equal to 20% of the PHTP and the reactor core <u>is not</u> critical, the minimum temperature requirement for the RV must be at least as high as the ART 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 ART 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 ART 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 ART 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). 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 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.

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 22) 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 23). 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 SUMMARY OF RESULTS

The results of this task provide a basis for the relocation of RCS P-T limits, LTOP setpoints, RV Surveillance and Neutron Fluence reporting requirements from the TSs to another controlled document called a PTLR.

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.

A generic approach for the relocation of the detailed information for the affected LCOs from the TSs based on GL 96-03 was used. A generic document, called an RCS PTLR, which contains the detailed information needed to comply with relocating the LCOs from the TSs can be developed based on information in this topical report.

An example PTLR and a sample TSs "mark-up" are provided in Appendices A & B & C. The example PTLR contains typical LCOs for RCS P-T limits and LTOP requirements for CE NSSS designs and can be tailored for plant specific submittals. The sample TSs "mark-up" is provided for illustrative purposes only. CEOG utilities must prepare plant-specific "mark-ups" of their current TSs for their individual submittals.

In conclusion, this report provides a referenceable generic basis for the creation of plant-specific PTLRs.

9.0 REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, Fracture Toughness Requirements, Federal Register, Vol. 60, No. 243, December 19, 1995, page 65474.

2. U.S. Nuclear Regulatory Commission, Standard Review Plan 5.2.2, Overpressure Protection, Revision 2, November 1988.

3. NRC GL 96-03, Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits, January 31, 1996.

4. Deleted.

5. "The ROCS and DIT Computer Codes for Nuclear Design", CENPD-266-P-A, April 1983.

6. "C-E Methodology for Core Design Containing Gadolinia-Urania Burnable Absorbers", CENPD-275, Rev. 1-P-A, May 1988.

7. "Methodology for Core Designs Containing Erbium Burnable Absorbers", CENPD-382-P-A, August 1993.

8. J.F. Carew, et. al, "Pressure Vessel Fluence Calculation Benchmark Problems and Solutions", NUREG/CR-6115 (Draft).

9. ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure", 1986 Edition.

10. ASME Boiler and Pressure Vessel Code, 1995 Edition and addenda through the 1996 Addenda, Section XI, Appendix A, "Analysis of Flaws" and Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

11. Cases of ASME Boiler and Pressure Vessel Code, Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, dated February 26, 1996.

12. U. S. Nuclear Regulatory Commission, Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures", Revision 1, November 1988.

13. ANSI/ISA-S67.04, Part I-1994, 'Setpoints for Nuclear Safety-Related Instrumentation, approved August 24, 1995.

14. WRCB 175 (Welding Research Council Bulletin 175), "PVRC Recommendations on Toughness Requirements for Ferritic Materials", August 1972.

15. U.S. Nuclear Regulatory Commission, Standard Review Plan 5.3.2, "Pressure-Temperature Limits", Rev. 1, July 1981.

16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

17. Title 10, Code of Federal Regulations, Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, Federal Register, Vol. 60, No. 243, December 19, 1995, page 65476.

18. ASTM E185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", American Society for Testing and Materials, Annual Book of Standards, Volume 12.02. (Applicable years 1966 – 1973).

19. "Heat Transfer, A Basic Approach", M. Cecati Ozisik, McGraw Hill Book Company, 1985.

20. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.105, Revision 3, December 1999, "Instrument Setpoint For Safety Related Systems".

21. OPPD Letter to NRC, LIC-99-0072, dated 8/6/99, withdrawal of application for Amendment of Operating License (contained within CE Memo O-PENG-99-013, Rev. 0, dated 8/27/99).

22. "Application of Reactor Vessel Surveillance Data for Embrittlement Management," CEN-405-P, Revision 03, dated September 1996.

23. "Improved Embrittlement Correlations for Reactor Pressure Vessel Steels," NUREG/CR-6551, dated November 1998.

APPENDIX A

(TOTAL PAGES: 16)

EXAMPLE OF

RCS PRESSURE AND TEMPERATURE LIMITS REPORT

June 2000

[NAME] UNIT ***[X]***

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

[DATE]

Not to be used for operation.

For illustration only.

[Note: This example is formatted so that the "plant specific information" or "optional" items are in ***Bold/Italic*** font and are enclosed in square brackets as shown on this page.]

Table of Contents

LIST OF FIGURES

1.0 INTRODUCTION

This PTLR for ***[NAME]*** Unit ***[X]*** contains Pressure-Temperature (P-T) limits corresponding to ***[Z]*** Effective Full Power Years (EFPY) of operation. In addition, this report contains Low Temperature Overpressure Protection (LTOP) specific requirements which have been developed to protect the P-T limits from being exceeded during the limiting LTOP event.

The Technical Specifications affected by this report are listed below and are separated into the appropriate category: P-T limits or LTOP requirements.

2.0 GL 96-03 PROVISION REQUIREMENTS

2.1 Neutron Fluence Values

The reactor vessel beltline neutron fluence has been calculated for the critical locations in accordance with the general methodologies as described in Section 1.0 of Reference 3.2. The following discussion gives the results of the fluence calculation followed by the details of the calculational analysis for the ***[NAME]*** Unit ***[X]***.

The peak value(s) of neutron fluence (E > 1 MeV) at the vessel clad interface used as input to the Adjusted Reference Temperature (ART) calculations for ***[NAME]*** Unit ***[X]*** corresponding to ***[locations on the vessel]*** for ***[Z]*** effective full power years (EFPY) is ***[$3.6x10^{19}$]*** neutrons per square centimeter (n/cm^2) with an associated uncertainty of ± ***[....]***.

2.1.1 Input Data

2.1.1.1 Materials and Geometry

[Details of materials and geometry in accordance with section 1.1.1 of Ref. 3.2]

2.1.1.2 Cross Sections

[Details of the cross sections used in accordance with section 1.1.2 of Ref. 3.2]

2.1.1.2.1 Multi-group Libraries

[Details of the multi group cross section library in accordance with section 1.1.2.1 of Ref. 3.2]

2.1.1.2.2 Construction of the Multi-Group Library

[Details of the construction of the multi-group library in accordance with section 1.1.2.2 of Ref. 3.2]

2.1.2 Core Neutron Source

[Details of the core neutron source in accordance with section 1.2 of Ref. 3.2]

2.1.3 Fluence Calculation

2.1.3.1 Transport Calculation

[Details of the transport calculation used in accordance with section 1.3.1 of Ref. 3.2]

2.1.3.2 Synthesis of the 3-D Fluence

[Details of the 3-D fluence synthesis in accordance with section 1.3.2 of Ref. 3.2]

2.1.3.3 Cavity Fluence Calculations

[Details of the cavity fluence calculation in accordance with section 1.3.3 of Ref. 3.2]

2.1.4 Methodology Qualification and Uncertainty Estimates

[Details of the methodology qualification and uncertainty estimates used in accordance with section 1.4 of Ref. 3.2]

2.1.4.1 Analytic Uncertainty Analysis

[Details of the analytical uncertainty analysis in accordance with section 1.4.1 of Ref. 3.2]

2.1.4.2 Comparison with Benchmark and Plant-Specific Measurements

[Details of the comparisons with benchmark and plant-specific measurements in accordance with section 1.4.2 of Ref. 3.2]

2.1.4.2.1 Operating Reactor Measurements

[Details of the reactor measurements comparisons with calculation in accordance with section 1.4.2.1 of Ref. 3.2]

2.1.4.2.2 Pressure Vessel Simulator Measurements

[Details of the pressure vessel simulator benchmark analyses performed in accordance with section 1.4.2.2 of Ref. 3.2]

2.1.4.2.3 Calculational Benchmarks

[Details of the calculational benchmark for methods qualification in accordance with section 1.4.2.3 of Ref. 3.2]

2.1.4.3 Overall Bias and Uncertainty

[Details of the overall bias and uncertainty analysis in accordance with section 1.4.3 of Ref. 3.2]

2.2 Reactor Vessel Surveillance Program

The reactor vessel surveillance program and the surveillance capsule withdrawal are described in Section 2, Reference 3.2 and Reference 3.***[q .. plant specific details including withdrawal schedule reference]***. The reports describing the post-irradiation evaluation of the surveillance capsules are contained in Reference 3.***[s .. post-irradiation evaluation reference]***

2.3 LTOP System Limits

The LTOP requirements have been developed by making a comparison between the peak transient pressures and the appropriate Appendix G pressure-temperature limit curves. The acceptability criterion regarding each particular transient is that the peak transient pressure does not exceed the applicable Appendix G pressure limits. The requirements for LTOP have been established based on NRC-accepted methodologies and are described in Section 3.0, Reference 3.2 and specified in the Bases Section for Technical Specification ***[A.B.C]***, Reference 3.3.

Several Technical Specification Limiting Conditions for Operation (LCOs) ensure adequate LTOP. However, pursuant to the guidelines of GL 96-03, only the pressure/temperature (P/T) limit curves and LTOP system limits may be relocated to a plant-specific PTLR. The other LTOP limitations must remain in the Technical Specifications. Accordingly, only the technical specifications on P/T limits ***[3.4.9.1]*** and LTOP limits ***[3.4.13]*** are addressed here.

2.3.1 <u>Pressure/Temperature Limits - Reactor Coolant System ([*LCO 3.4.9.1*])</u>

2.3.1.1 The RCS (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures ***[4-1, 4-2, and 43-]*** during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. See Section 2.5 below for details.

[Details of the P/T limit development methodology in accordance with Section 5.0 of Reference 3.2]

2.3.2 <u>Reactor Coolant System Power Operated Relief Valves ([*LCO 3.4.13*])</u>

2.3.2.1 The setpoints for the power operated relief valves shall be as follows:

a. A setpoint of less than or equal to ***[350 psia]*** shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to ***[215°F]*** and
2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to ***[193°F]***.

b. A setpoint of less than or equal to ***[530 psia]*** shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than ***[215°F]*** and less than or equal to the LTOP Enable Temperature for cooldown.
2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to ***[193°F]*** and less than or equal to the LTOP Enable Temperature for heatup.

[Details of the setpoint development methodology in accordance with Section 3.0 of Reference 3.2]

2.3.5.2 The LTOP Enable Temperatures are defined as follows:

a. The LTOP Enable Temperature for heatup is ***[304°F]***.

b.

The LTOP enable temperatures are determined using guidance provided in Branch Technical Position RSB 5-2.

2.4 Beltline Material Adjusted Reference Temperature (ART)

The calculation of the adjusted reference temperature (ART) for the beltline region has been performed using the NRC-accepted methodologies as described in Section 4.0, Reference 3.2. Application of Surveillance Data ***[was/was not]*** used to refine the chemistry factor and the margin term (see Section 2.7 below).

The limiting ART values in the beltline region for the ***[NAME]*** Unit ***[X]*** corresponding to ***[Z]*** Effective Full Power Years (EFPY) for the 1/4t and 3/4t locations are:

Location	ART	Material
1/4t	***[xxx °F]***	***[... Limiting Plate or Weld Material Identification ...]***
3/4t	***[xxx °F]***	***[... Limiting Plate or Weld Material Identification ...]***

The RT_{PTS} value for ***[NAME]*** Unit ***[X]*** which is calculated in accordance with 10 CFR 50.61 is ***[xxx °F]*** which corresponds to ***[Limiting Plate or Weld Identifier]***. Application of Surveillance Data ***[was/was not]*** used to refine the chemistry factor and the margin term (see Section 2.7).

2.5 Pressure-Temperature Limits Using Limiting ART in the P-T Curve Calculation

The limits for ***[LCO 3.4.9.1]*** are presented in the subsection that follows. The analytical methods used to develop the RCS pressure-temperature limits are based on NRC-accepted methodologies and discussed in Section 5.0 of Reference 3.2. The methodology is also documented in the Bases for Technical Specification ***[A.B.C]***.

The RCS PRESSURE-TEMPERATURE LIMITS REPORT will be updated prior to exceeding the RT_{NDT} utilized to develop the current heatup and cooldown curves. The RCS PRESSURE-TEMPERATURE LIMITS REPORT, including any revisions or supplements thereto, shall be provided, upon issuance of new heatup and cooldown curves to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

2.5.1 RCS Pressure and Temperature (P/T) Limits ***([LCO 3.4.9.1])***

2.5.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup of ***[75]***°F in any 1-hour period***, as shown in Figure 4-1.***

b. A maximum cooldown rate ***as shown in Figures 4-2 and 4-3.***

c. A maximum temperature change of $\leq$ 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.5.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by ***Figures 4-1, 4-2 and 4-3.***

2.6 Minimum Temperature Requirements in the P-T Curves

The minimum temperature requirements specified in Appendix G to 10 CFR 50 are applied to the P/T curves using the NRC-accepted methodologies as described in Section 6.0 of Reference 3.2.

The minimum temperature values applied to the P/T curves for ***[NAME]*** Unit ***[X]*** corresponding to ***[Z]*** Effective Full Power Years (EFPY) are:

Location	Min Temperature
BoltUp	***[80°F]***
Hydrotest	***[per Figure 4-1]***
Lowest Service	***[190 °F]***

The lowest service temperature is established based on the limiting RT_{NDT} for the reactor coolant pumps.

2.7 Application of Surveillance Data to ART Calculations

Post-irradiation surveillance capsule test results for ***[NAME]*** Unit ***[X]*** are given in ***[Reference 3.s]***. The test results ***[do/do not]*** meet the credibility criteria of Regulatory Guide 1.99 Revision 2. ***[The criteria were met as follows:***

a) the surveillance program plate or weld duplicates the controlling reactor vessel beltline material in terms of ART;

b) Charpy data scatter does not cause ambiguity in the determination of the 30 ft-lb shift;

c) the measured shifts are consistent with the predicted shifts;

d) the capsule irradiation temperature is comparable to that of the vessel; and

e) correlation monitor data [are/are not] available and are consistent with the known data for that material.

The data supporting the credibility analysis are presented in [reference].]

[In the case where sister vessel surveillance data are available for use, the preceding should be supplemented as indicated under Section 7.0 of Reference 3.2. The supplemental information should address differences between the two sister plants in terms of irradiation environment and establish the applicability of the data.]

The credible surveillance data *[were/were not]* used to refine the chemistry factor and the margin term. *[The process for applying the credible surveillance data is described under Section 7.0 of Reference 3.2 in the Methodology and follows that prescribed in Position 2.1 of Regulatory Guide 1.99, Revision 2. The data used and the calculations performed are given below:*

Report	*Capsule ID*	*Fluence*	*Shift*	*Fluence Factor, f*	*$(f)^2$*	*(f x shift)*
...	...	...	...	...	..	...
...	...	...	...	...	..	...

Refined Chemistry Factor, CF(R)= $$\frac{\Sigma(f \times shift)}{\Sigma(f)^2}$$

σ_Δ *= (17 or 28)°F*

Refined σ_Δ *= (17 or 28)°F /2*

Refined ART = Initial Rtndt + CF(R) x f + $2\sqrt{(\frac{\sigma_\Delta}{2})^2 + \sigma_1^2}$ *]*

3.0 REFERENCES

3.1 NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits", January 31, 1996.

3.2 CE NPSD-683P, Rev 05, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP requirements from the Technical Specifications," September 1999.

3.3 ***Tech Spec A.B.C for [Name] Unit [X] ...***

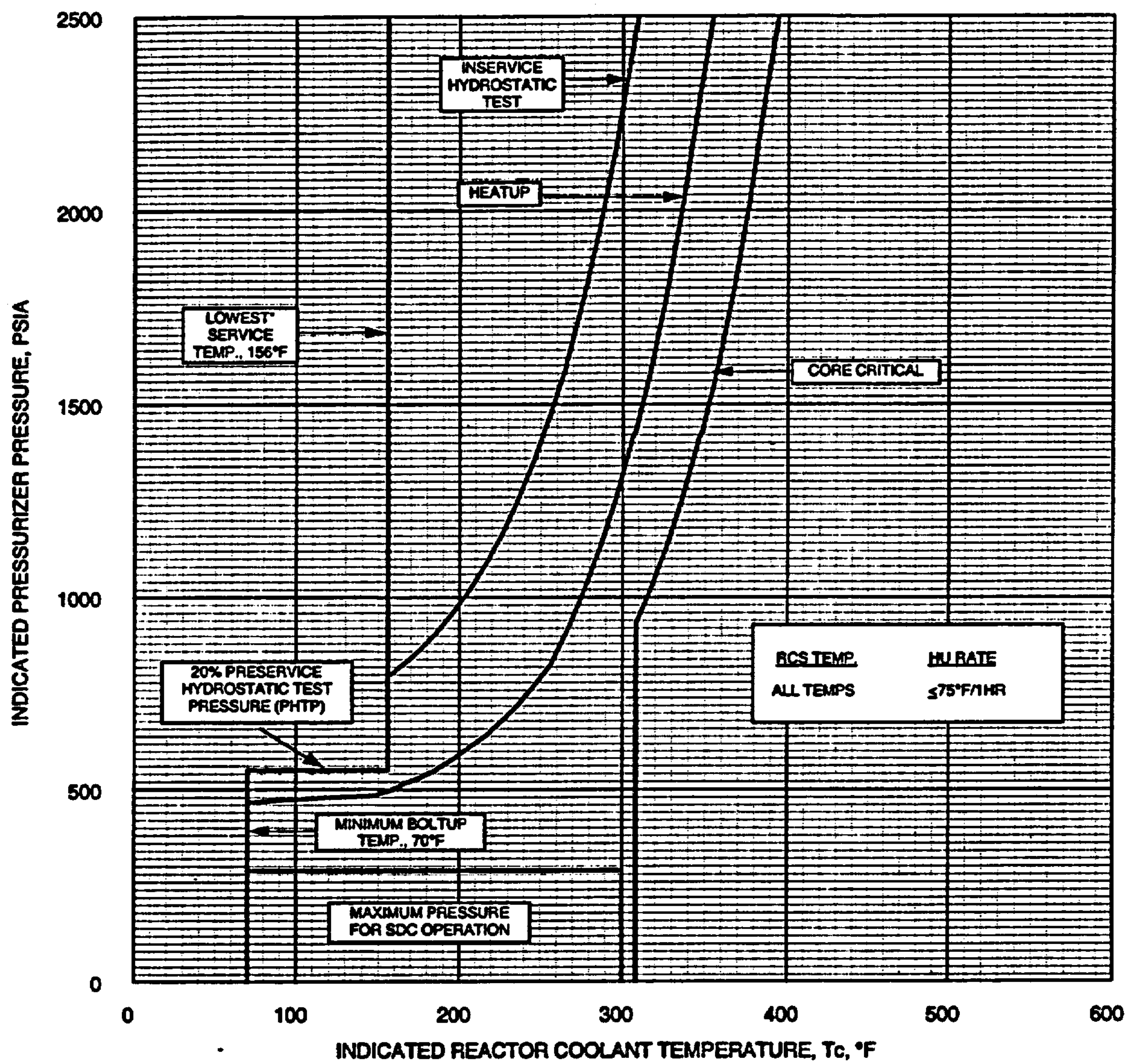
[3.q IF not in Tech Spec ... Reference for Plant Specific Surveillance Capsule Withdrawal Schedule]

[3.s Reference for post-irradiation evaluation of surveillance capsules]

[3.z Reference for Fluence value modification]

FIGURE 4-1

[NAME] UNIT [A] P/T LIMITS, [ ] EFPY
HEATUP AND CORE CRITICAL

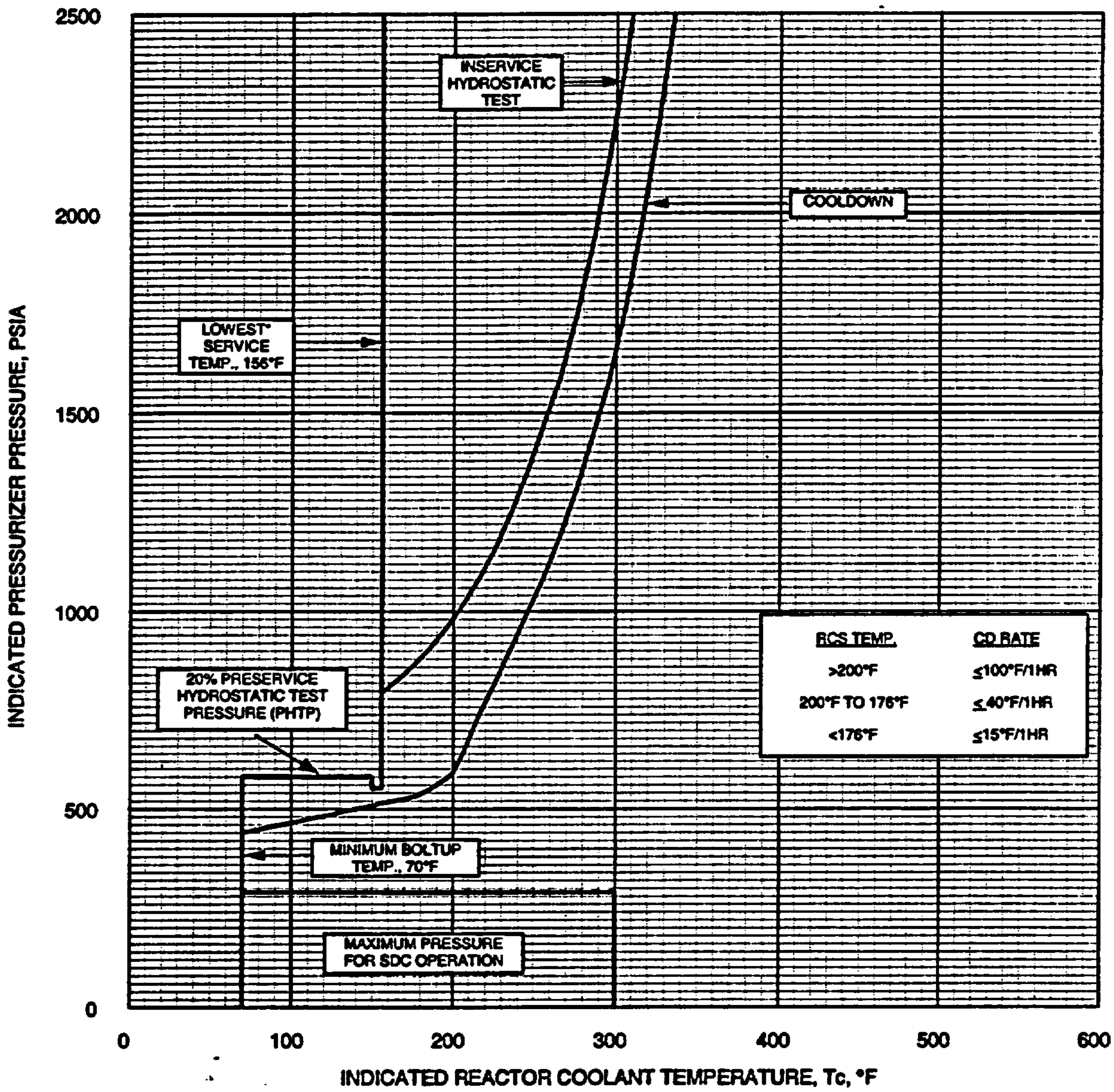


*This line is defined as the more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

[NAME] UNIT [A] AMENDMENT NO. [Y]

FIGURE 4-2

[NAME] UNIT [A] P/T LIMITS, [ ] EFPY
COOLDOWN AND INSERVICE TEST

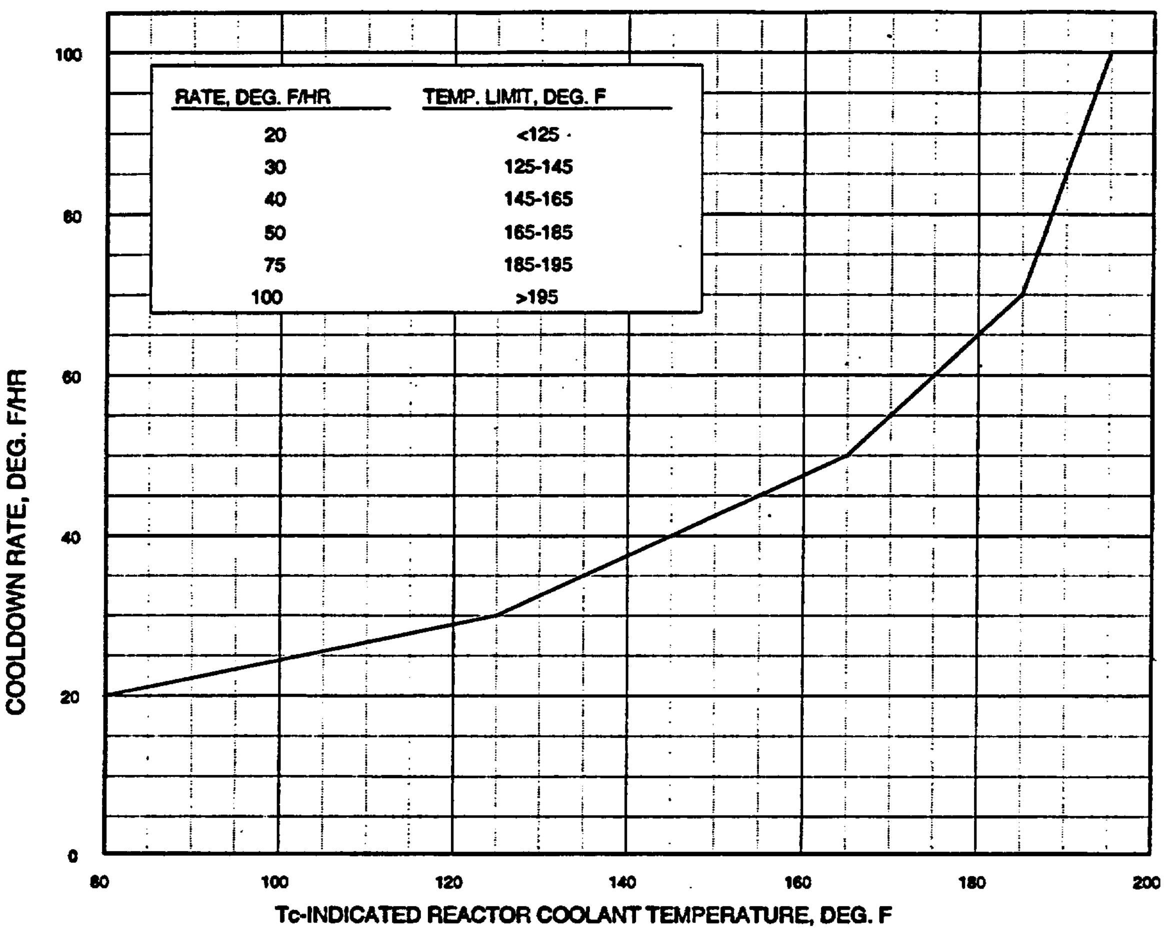


*This line is defined as the more conservative of either the Lowest Service Temperature or the minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of PHTP.

[NAME] UNIT [A] AMENDMENT NO. [Y]

FIGURE 4-3

[NAME] UNIT [A] P/T LIMITS, [ ] EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100 DEG. F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 195 DEG. F

[NAME] UNIT [A] AMENDMENT NO. [Y]

APPENDIX B

(TOTAL PAGES: 33)

EXAMPLE OF MODIFIED

TECHNICAL SPECIFICATIONS

Note: The Technical Specification markups presented in this appendix are for information purposes only and are not for formal review. The intent of this topical report is not to propose changes to the Technical Specifications. Technical Specification changes, as needed, will be submitted on a plant specific basis.

INDEX

DEFINITIONS

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

1.16 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is $< 304°F$ during heatup or $\leq 281°F$ during cooldown and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 1.75 square inches.

Replace with

less than the respective LTOP Enable Temperature specified in the RCS Pressure-Temperature Limits Report.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculations of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Sample point locations.

INSERT A

PRESSURE TEMPERATURE LIMITS REPORT (PTLR)

1.26 The PTLR is the unit specific document that provides the reactor vessel P-T limits, including heatup and cooldown rates, and LTOP setpoints for the current reactor vessel fluence period. These P-T limits shall be determined for fluence period of effective full-power years (EFPYs) in accordance with Specification 6.9.1.12. Plant operation within these operating limits is addressed in Specification [3/4.4.9] [RCS Pressure/Temperature Limits] and Specification [3.4.13] [Low Temperature Overpressure Protection].

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

INSERT (B)

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.32 3 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.33 4 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.34 5 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.35 6 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.36 7 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.37 8 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. A least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

replace with Insert B

* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

High Pressure Header	Auxiliary Header
HCV-3616	HCV-3617
HCV-3626	HCV-3627
HCV-3636	HCV-3637
HCV-3646	HCV-3647

INSERT B

the maximum allowable rates for single HPSI pump in operation in the PTLR

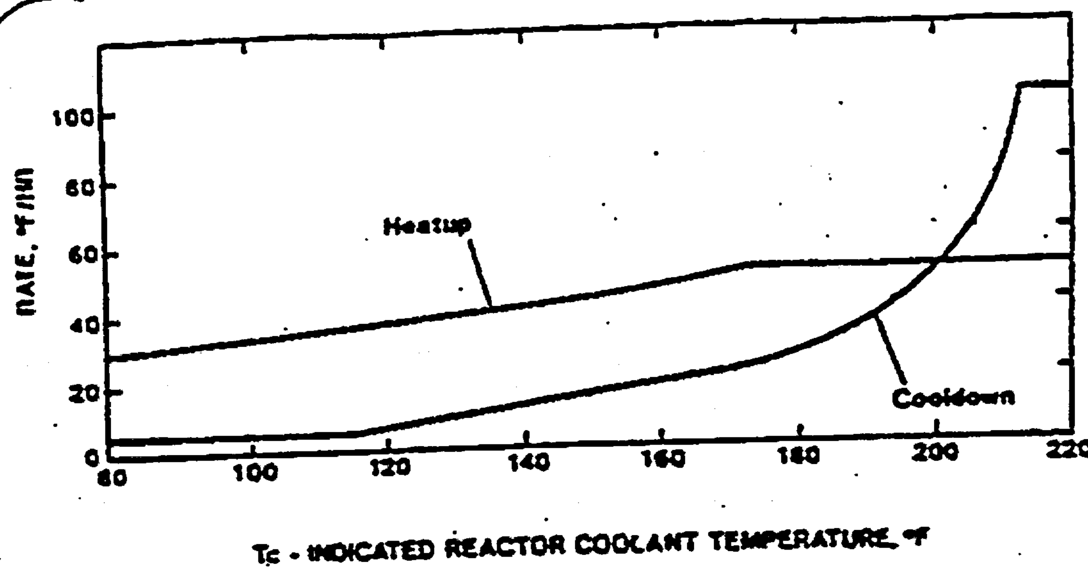


FIGURE 3.1-1b
MAXIMUM ALLOWABLE HEATUP AND COOLDOWN RATES,
SINGLE HPSI PUMP IN OPERATION

Remove to PTLR

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to the Inservice Testing Program.

Replace with Insert B

* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

High Pressure Header	Auxiliary Header
HCV-3616	HCV-3617
HCV-3626	HCV-3627
HCV-3636	HCV-3637
HCV-3646	HCV-3647

<u>INSERT B</u>

the maximum allowable rates for single HPSI pump in operation in the PTLR

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2a, 3.4-2b and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

Replace with Insert C

APPLICABILITY: At all times *#

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the following 30 hours in accordance with Figures 3.4-2b and 3.4-3.

LCO 3.4.9.1

*When the flow path from the RWT to the RCS via a single HPSI pump is established per 3.1.2.3, the heatup and cooldown rates shall be established ~~in accordance with Fig. 3.1-1b.~~ as indicated in the PTLR.

#During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F.

INSERT C

The combination of RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits specified in the RCS PRESSURE-TEMPERATURE LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

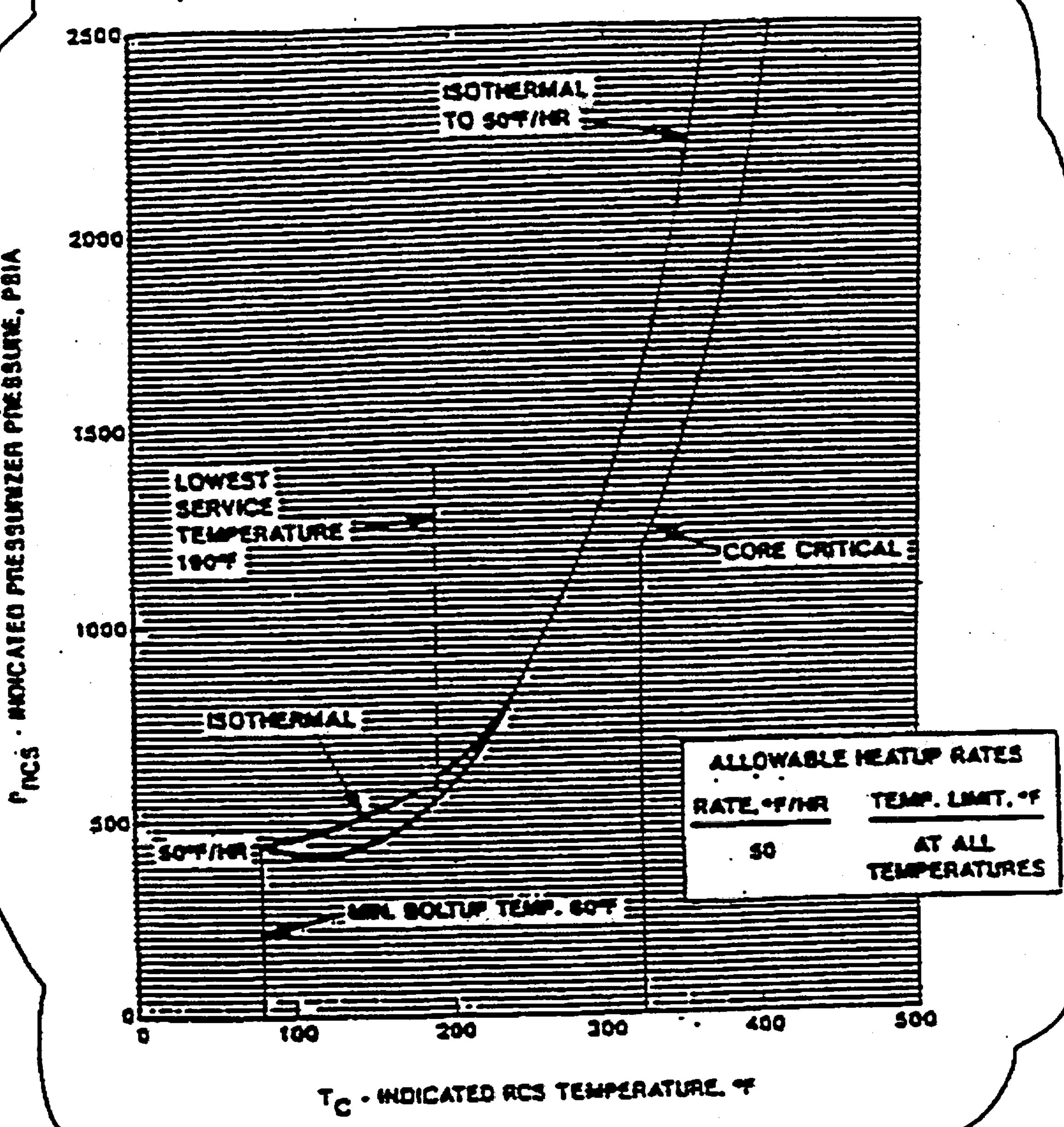
c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update ~~Figures 3.4-2a, 3.4-2b and 3.4-3~~

the PTLR

REMOVE TO PTLR

FIGURE 3.4-2a

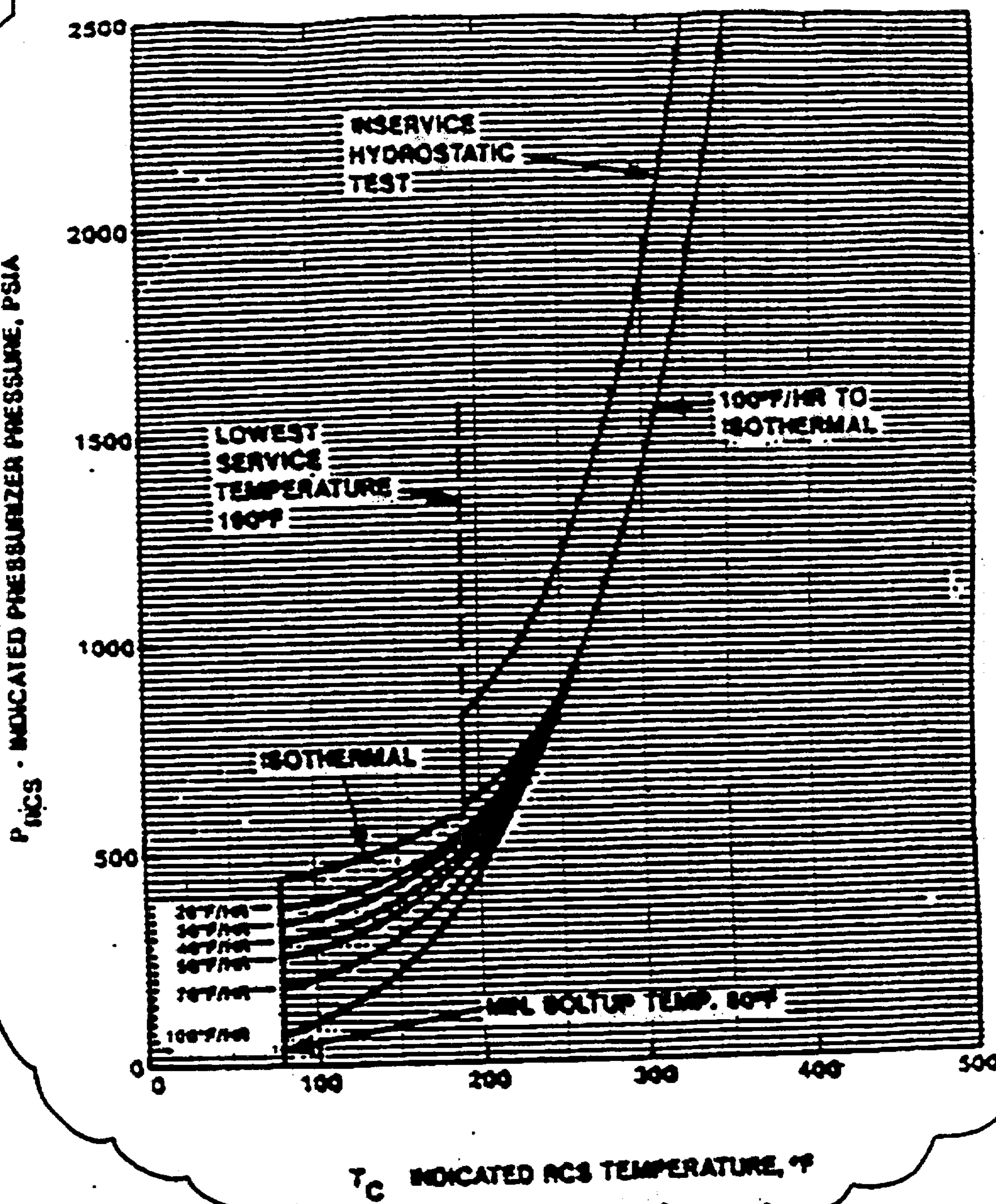
[NAME] UNIT [X] P/T LIMITS, [ ] EFPY
HEATUP AND CORE CRITICAL



3/4 4-23a

FIGURE 3.4-2b

[NAME] UNIT [A] P/T LIMITS, [ ] EFPY
COOLDOWN AND INSERVICE TEST

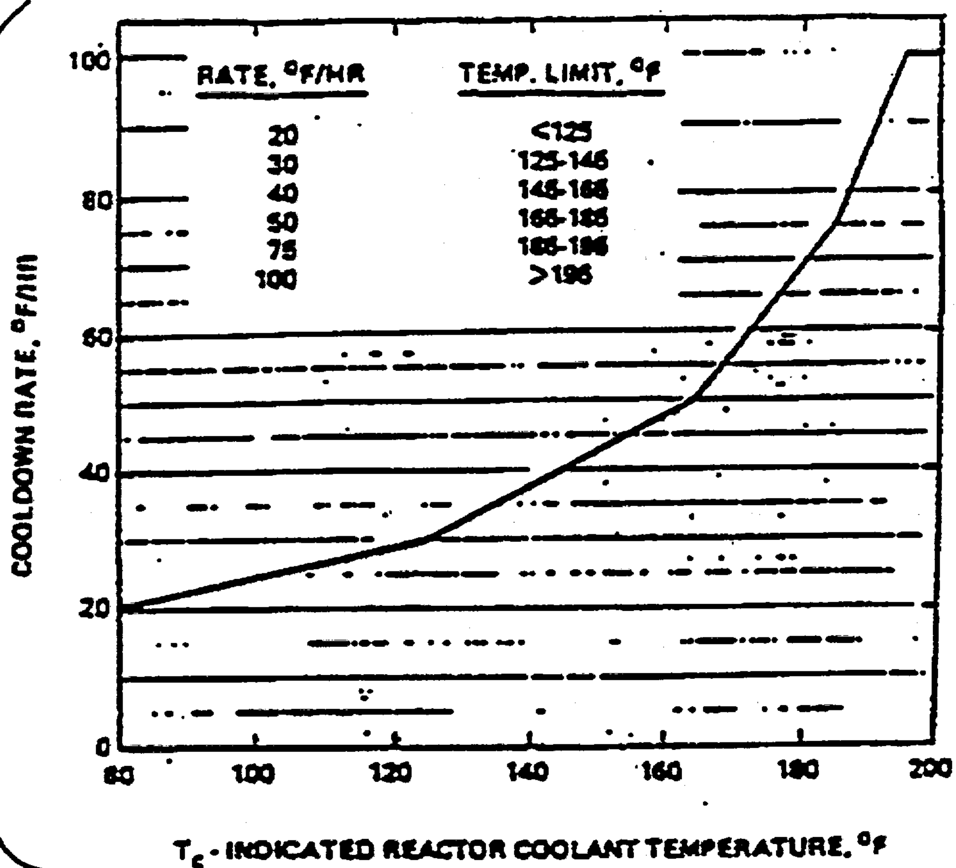


REMOVE TO PTLR

3/4 4-4-23b

FIGURE 3.4-3

[NAME] UNIT [A], [ ] EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 195°F

REMOVE TO PTLR

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

in accordance with those Specified in the PTLR

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to 215°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 193°F.

b. A setpoint of less than or equal to 530 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than 215°F and less than or equal to 281°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 193°F and less than or equal to 304°F.

REMOVE TO PTLR

APPLICABILITY: MODES 4# and 5*.

ACTION:

a. With less than two PORVs OPERABLE and while at Hot Shutdown during a planned cooldown, both PORVs will be returned to OPERABLE status prior to entering the applicable MODE unless:

1. The repairs cannot be accomplished within 24 hours or the repairs cannot be performed under hot conditions, or

2. Another action statement requires cooldown, or

3. Plant and personnel safety requires cooldown to Cold Shutdown with extreme caution.

b. With less than two PORVs OPERABLE while in COLD SHUTDOWN, both PORVs will be returned to OPERABLE status prior to startup.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 The PORVs shall be verified OPERABLE by:

a. Verifying the isolation valves are open when the PORVs are reset to the low temperature mode of operation.

b. Performance of a CHANNEL FUNCTIONAL TEST of the Reactor Coolant System overpressurization protection system circuitry up to and including the relief valve solenoids once per refueling outage.

c. Performance of a CHANNEL CALIBRATION of the pressurizer pressure sensing channels once per 18 months.

less than or equal to the LTOP Enable Temperature specified in RCS Pressure-Temperature Limits Report.

#Reactor Coolant System cold leg temperature below 304°F.

*PORVs are not required below 140°F when RCS does not have pressure boundary integrity.

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP - STARTING

LIMITING CONDITION FOR OPERATION

3.4.14 If the steam generator temperature exceeds the primary temperature by more than 30°F, the first idle reactor coolant pump shall not be started.

APPLICABILITY: MODES 4# and 5.

ACTION:

If a reactor coolant pump is started when the steam generator temperature exceeds primary temperature by more than 30°F, evaluate the subsequent transient to determine compliance with Specification 3.4.9.1.

SURVEILLANCE REQUIREMENTS

4.4.14 Prior to starting a reactor coolant pump, verify that the steam generator temperature does not exceed primary temperature by more than 30°F.

or equal to the LTOP Enable Temperature specified in the RCS Pressure - Temperature Limits Report (PTLR).

#Reactor Coolant System Cold Leg Temperature is less than 304°F.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position - Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor - F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations - Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A), Rev. 0 and Supplement 1 through 5, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Rev. 0 dated June 1975, Supplement 1 dated September 1976, Supplement 2 dated December 1980, Supplement 3 dated September 1981, Supplement 4 dated December 1986, Supplement 5 dated February 1987.

4. ANF-84-73(P), Rev. 3, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuel Corporation, dated May 1988.

5. XN-NF-82-21(A), Rev. 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, dated September 1983.

6. ANF-84-93(A), Rev. 0 and Supplement 1, "Steamline Break Methodology for PWR's," Advanced Nuclear Fuels Corporation, Rev. 0 dated March 1989, Supplement 1 dated March 1989.

CORE OPERATING LIMITS REPORT (continued)

7. XN-75-32(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, dated October 1983.

8. XN-NF-82-49(A), Rev. 1 and Supplement 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, Rev. 1 dated April 1989, Supplement 1 dated December 1994.

9. XN-NF-78-44(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, dated October 1983.

10. XN-NF-621(A), Rev. 1, "Exxon Nuclear DNB Correlation of PWR Fuel Design," Exxon Nuclear Company, dated September 1983.

11. EXEM PWR Large Break LOCA Evaluation Model as defined by:

    a) XN-NF-82-20(A), Rev. 1 and Supplements 1 through 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, all dated January 1990.

    b) XN-NF-82-07(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, dated November 1982.

    c) XN-NF-81-58(A), Rev. 2 and Supplements 1 through 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Rev. 2 and Supplement 1 and 2 dated March 1984, Supplements 3 and 4 dated June 1990.

    d) XN-NF-85-16(A), Volume 1 through Supplement 3; Volume 2, Rev. 1 and Supplement 1, "PWR 17x17 Fuel Cooling Tests Program," Exxon Nuclear Company, all dated February 1990.

    e) XN-NF-85-105(A), Rev. 0 and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company, all dated January 1990.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

INSERT D

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

INSERT D

6.9.1.12 REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, LTOP, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for Specification [3/4.4.9 RCS Pressure/Temperature Limits] and Specification [3.4.13 Low Temperature Overpressure Protection].

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document(s):

   1. NRC letter dated [date], [Title of Letter], and

   2. CE NPSD-683-P, Rev. 05, The Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Setpoints from the Technical specifications, July, 2000.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period of EFPYs and for any revision or supplement thereto.

Remove and replace with Insert E

3/4.4.13 POWER OPERATED RELIEF VALVES and 3/4.4.14 REACTOR COOLANT PUMP - STARTING

The low temperature overpressure protection system (LTOP) is designed to prevent RCS overpressurization above the 10 CFR 50 Appendix G operating limit curves (Figures 3.4-2a and 3.4-2b) at RCS temperatures at or below 304°F during heatup and 281°F during cooldown. The LTOP system is based on the use of the pressurizer power-operated relief valves (PORVs) and the implementation of administrative and operational controls.

The PORVs aligned to the RCS with the low pressure setpoints of 550 and 530 psia, restrictions on RCP starts, limitations on heatup and cooldown rates, and disabling of non-essential components provide assurance that Appendix G P/T limits will not be exceeded during normal operation or design basis overpressurization events due to mass or energy addition to the RCS. The LTOP system APPLICABILITY, ACTIONS, and SURVEILLANCE REQUIREMENTS are consistent with the resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," pursuant to Generic Letter 90-06.

replace with "contained in the PTLR"

3/4.4.15 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

INSERT E

found in the PTLR, at or below the LTOP enable temperatures as specified in the PTLR.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

The limitations on HPSI pump operability when the RCS temperature is ≤ 270°F and ≤ 236°F, and the associated Surveillance Requirements provide additional administrative assurance that the pressure/temperature limits ~~(Figures 3.4-2a and 3.4-2b)~~ will not be exceeded during a mass addition transient mitigated by a single PORV. A limit on the maximum number of operable HPSI pumps is not necessary when the pressurizer manway cover or the reactor vessel head is removed.

Remove and replace
with "contained in the PTLR"

B 3/4 5-1

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturaction pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take correction action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside surface and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location than at the outside surface location, the inside surface flaw may be more limiting. Consequently, for the heatup analysis, both the inside surface and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface.

Since neutron irradiation damage is also greater at the inside surface, the inside surface flaw location is the limiting location during cooldown. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

→ INSERT F

INSERT F

The PTLR contains the pressure-temperature limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The curves are developed based upon the NRC-approved methodology contained in the CEOG Topical Report CE NPSD-683, Rev. 05, dated ...

Remove and replace with "in the PTLP"

The heatup and cooldown limit curves ~~(Figures 3.4-2a and 3.4-2b)~~ are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50°F/hr and for any cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the applicable service period.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be calculated based upon the fluence. The heatup and cooldown limit curves ~~shown on Figures 3.4-2a and 3.4-2b~~ include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material surveillance specimens installed near the inside wall of the reactor vessel in the core area. The capsules are scheduled for removal at times that correspond to key accumulated fluence levels within the vessel through the end of life. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, measured ΔRT_{NDT} for surveillance samples can be applied with confidence to the corresponding material in the reactor vessel wall. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines ~~shown on Figures 3.4-2a and 3.4-2b~~ for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been estimated to be 90°F. The Lowest Service Temperature limit line shown on ~~Figures 3.4-2a and 3.4-2b~~ is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

remove and replace with "the P-T limit curves in the PTLP"

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	NI %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG[1]	50 FT-LB/35 MIL TEMP F TRANS[1,2]	RTNDT[4] F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS[3]
Vessel Flange Forging	C-1-1	A508Cl.2	-	-	.008	+20	+70	+90	+30	133	86
Bottom Head Plate	C-10-1	A533BCl.1	-	-	.010	-40	+42	+62	+2	120	78
Bottom Head Plate	C-9-2	A533BCl.1	-	-	.011	-40	-18	+2	-40	146	95
Bottom Head Plate	C-9-3	A533BCl.1	-	-	.013	-70	-20	0	-60	148	96
Bottom Head Plate	C-9-1	A533BCl.1	-	-	.011	-30	+10	+30	-30	138	90
Inlet Nozzle	C-4-3	A508Cl.2	-	-	.005	0	0	+20	0	111	72
Inlet Nozzle	C-4-2	A508Cl.2	-	-	.004	0	+20	+40	0	146	95
Inlet Nozzle	C-4-1	A508Cl.2	-	-	.005	+10	-25	-5	10	144	94
Inlet Nozzle	C-4-4	A508Cl.2	-	-	.004	0	+10	+30	0	139	90
Inlet Nozzle Ext.	C-16-3	A508Cl.2	-	-	.001	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-2	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-1	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-4	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90

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TABLE B 3/4.4.1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	NI %	P %	NDTT F	50 FT-LB/35 MIL TEMP F		RTNDT[3] F	MIN. UPPER SHELF FT-LB	
							LONG[1]	TRANS[1,2]		LONG	TRANS [4]
Outlet Nozzle	C-3-1	A508C1.2	-	-	.009	+10	+88	+108	+48	119	77
Outlet Nozzle	C-3-2	A508C1.2	-	-	.010	-20	+92	+112	+52	111	72
Outlet Nozzle Ext.	C-17-1	A508C1.2	-	-	.013	+20	-	-	+28[6]	126	82
Outlet Nozzle Ext.	C-17-2	A508C1.2	-	-	.013	+20	-	-	+28[6]	126	82
Upper Shell Plate	C-6-3	A533BC1.1	-	-	.011	-10	+30	+50	-10	129	84
Upper Shell Plate	C-6-2	A533BC1.1	-	-	.010	-30	+45	+65	+5	123	80
Upper Shell Plate	C-6-1	A533BC1.1	-	-	.012	+10	+42	+62	+10	105	68
Inter. Shell Plate	C-7-1	A533BC1.1	0.11	0.64	0.004	0	+28	+48	0	126	82
Inter. Shell Plate	C-7-2	A533BC1.1	0.11	0.64	0.004	-30	+30	+50	-10	131	85
Inter. Shell Plate	C-7-3	A533BC1.1	0.11	0.58	0.004	-30	+50	+70	+10	117	76
Lower Shell Plate	C-8-3	A533BC1.1	0.12	0.58	0.004	0	+28	+46	0	136	88
Lower Shell Plate	C-8-1	A533BC1.1	0.15	0.56	0.008	-10	+60	+80	+20	126	82
Lower Shell Plate	C-8-2	A533BC1.1	0.15	0.57	0.006	0	+32	+52	20[7]	120	103[7]
Closure Head Flange	C-2	A508C1.2	-	-	.008	+20	-	-	+20[8]	143	93
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86

TABLE B 3/4.4.1 (Continued)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	NI %	P %	NDTT F	50 FT-LB/35 MIL TEMP F		RTNDT[4] F	MIN. UPPER SHELF FT-LB	
							LONG[1]	TRANS[1,2]		LONG	TRANS[7]
Closure Head Peels	C-21-1	A533BC1.1	-	-	.013	-10	0	+20	-10	138	90
Closure Head Peels	C-21-1	A533BC1.1	-	-	.013	-10	0	+20	-10	138	90
Closure Head Peels	C-21-2	A533BC1.1	-	-	.012	-30	+40	+60	0	133	86
Closure Head Peels	C-21-3	A533BC1.1	-	-	.013	-40	+36	+56	-4	129	84
Closure Head Dome	C-20-1	A533BC1.1	-	-	.014	-10	+44	+64	+4	105	68
Inter. Shell Long. Welds	2-203 A, B, C	A8746/34B009 Linde 124	0.19[9]	0.10[9]	.018	-	-	-	-56[8]	-	102.3[9]
Lower Shell Long. Welds[10]	3-203 A, B, C	305424 Linde 1092	0.28	0.63	.016	-60[8]	-	-	-60[8]	-	112[9]
Lower-to-Inter. Shell Seam Weld	9-203	90136 Linde 0091	0.23	0.11	.013	-60[7]	-	-38[7]	-60[7]	-	144[7]

Notes:
(1) Charpy 50 ft-lb and 35 mils lateral expansion index temperature (lower bound)
(2) Determined using Branch Technical Position MTEB 5-2, Section 1.1(3)(b)
(3) Determined by using Branch Technical Position MTEB 5-2 Section 1.2
(4) As per ASME B&PV Code, Section III, NB-2331
(5) Charpy test data either do not have lateral expansion value or the data are not legible. The reference temperature from Charpy test data was obtained by following MTEB Position 5.2, Section 1.1(4)
(6) Estimated based on generic data for C-E submerged arc welds ("Evaluation of Pressurized Thermal Shock Effects due to Small Break LOCA's with Loss of Feedwater for the Combustion Engineering NSSS," CEN-189, December 1981).
(7) Surveillance Program Data - Average USE
(8) Estimate based on generic data for CE submerged arc welds (CE Reports CE-NP SD-906P, F-MECH-93-050).
(9) Initial Property for identical CE fabricated weld in the Beaver Valley Unit 1 Surveillance Program.
(10) Weld Chemistry is the mean of data from CE analysis and note 9.

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for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limiations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection program for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. This program is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

APPENDIX C

(TOTAL PAGES: 30)

EXAMPLE OF MODIFIED

TECHNICAL SPECIFICATIONS

IN THE FORMAT OF

CE STANDARD TECHNICAL SPECIFICATION (NUREG 1432)

Note: The Standard Technical Specification markups presented in this appendix are for information purposes only and are not for formal review. The intent of this topical report is not to propose changes to NUREG 1432. Technical Specification changes, as needed, will be done under processes defined by the NEI Technical Specification Task Force (TSTF).

Included:

1.1	Definitions: Pressure and Temperature Limits Report (PTLR)
3.4.3	RCS Pressure and Temperature (P/T) Limits
B.3.4.3	RCS Pressure and Temperature (P/T) Limits - Bases
3.4.12	Low Temperature Overpressure Protection (LTOP) System
B.3.4.12	Low Temperature Overpressure Protection (LTOP) System - Bases
5.6.6	Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

1.1 Definitions (continued)

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter [14, Initial Test Program] of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. -------------------	A.1 Restore parameter(s) to within limits.	30 minutes
Requirements of LCO not met in MODE 1, 2, 3, or 4.	AND	
	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5 with RCS pressure < [500] psig.	36 hours

(contin...

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed whenever this Condition is entered. -------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
	C.2	Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates within limits specified in the PTLR.	30 minutes

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, XI Appendix G (Ref. 3).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted,

(continued)

BASES

BACKGROUND (continued)

as necessary, based on the evaluation findings and the recommendations of Reference 3.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit includes the Reference 2 requirement that the limit be no less than 40°F above the heatup curve or the cooldown curve and not less than the minimum permissible temperature for the ISLH testing. However, the criticality limit is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) Analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

methodology for determining the P/T limits. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

The RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer.

These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

(continued)

BASES

LCO (continued)

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures.

ACTIONS

A.1 and A.2

Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because:

a. The RCS remained in an unacceptable P/T region for an extended period of increased stress; or

b. A sufficiently severe event caused entry into an unacceptable region.

Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

reduced pressure and temperature. With reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in MODE 3 within 6 hours and in MODE 5 with RCS pressure < [500] psig within 36 hours.

The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

The actions of this LCO, anytime other than in MODE 1, 2, 3, or 4, consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, at times accompanied by equipment failures, may also require additional actions from emergency operating procedures. Operation outside the P/T limits must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The Completion Time of "immediately" reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in a short period of time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The Completion Time of prior to entering MODE 4 forces the evaluation prior to entering a MODE where temperature and pressure can be significantly increased. The evaluation for a mild violation is possible within several days, but more severe violations may require special, event specific stress analyses or inspections.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that requires this SR be performed only during RCS system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. [NRC approved topical report that defines the methodology for determining the P/T limits].

2. 10 CFR 50, Appendix G.

(continued)

BASES

REFERENCES (continued)

3. ASME, Boiler and Pressure Vessel Code, Section III, XI
Appendix G.

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one high pressure safety injection (HPSI) pump and one charging pump capable of injecting into the RCS and the safety injection tanks (SITs) isolated, and:

a. Two OPERABLE ~~power operated~~ relief valves (PORVs) with lift settings ≤ [450] psig; or

b. The RCS depressurized and an RCS vent of ≥ [1.3] square inches.

Replace with as found in the PTLR

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ [285]°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

---------------------------NOTE---------------------------
SIT isolation is only required when SIT pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Two or more HPSI pumps capable of injecting into the RCS.	A.1	Initiate action to verify a maximum of one HPSI pump capable of injecting into the RCS.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more charging pumps capable of injecting into the RCS.	------------NOTE------------ Two charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. --------------------------	
	B.1 Initiate action to verify a maximum of one charging pump capable of injecting into the RCS.	Immediately
C. A SIT not isolated when SIT pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	C.1 Isolate affected SIT.	1 hour
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > [175]°F.	12 hours
	<u>OR</u>	
	D.2 Depressurize affected SIT to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
E. One required PORV inoperable in MODE 4.	E.1 Restore required PORV to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One required PORV inoperable in MODE 5 or 6.	F.1	Restore required PORV to OPERABLE status.	24 hours
G. Two required PORVs inoperable. OR Required Action and associated Completion Time of Condition A, [B,] D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, [B,] C, D, E, or F.	G.1	Depressurize RCS and establish RCS vent of ≥ [1.3] square inches.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of one HPSI pump is capable of injecting into the RCS.	12 hours
SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.3	-----------------NOTE----------------- Required to be performed when complying with LCO 3.4.12b. -------------------------------------- Verify each SIT is isolated.	12 hours
SR 3.4.12.4	Verify RCS vent ≥ [1.3] square inches is open.	12 hours for unlocked open vent valve(s) **AND** 31 days for locked open vent valve(s)
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	72 hours
SR 3.4.12.6	-----------------NOTE----------------- Not required to be performed until [12] hours after decreasing RCS cold leg temperature to ≤ [285]°F. -------------------------------------- Perform CHANNEL FUNCTIONAL TEST on each required PORV, excluding actuation.	Replace with "that found in the PTLR" 31 days
SR 3.4.12.7	Perform CHANNEL CALIBRATION on each required PORV actuation channel.	[18] months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for operational pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperatures. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3 requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one high pressure safety injection (HPSI) pump and one charging pump incapable of injection into the RCS and isolating the safety injection tanks (SITs). The pressure relief capacity requires either two OPERABLE redundant power operated relief valves (PORVs) or the RCS depressurized and an RCS vent of sufficient size. One PORV or the RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one [HPI or] charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings or an RCS vent of sufficient size. Two relief valves are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The actuation logic monitors RCS pressure and determines when the LTOP overpressure setting is approached. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The LCO presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits of the LCO ensures the P/T limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the system pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

(continued)

BASES

BACKGROUND (continued)

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the specified flow capacity, it requires removing a pressurizer safety valve, removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits during shutdown. In MODES 1, 2, and 3, and in MODE 4 with any RCS cold leg temperature exceeding [285]°F, the pressurizer safety valves prevent RCS pressure from exceeding the Reference 1 limits. At about [285]°F and below, overpressure prevention falls to the OPERABLE PORVs [or to a depressurized RCS and a sufficient sized RCS vent]. Each of these means has a limited overpressure relief capability.

Replace with the value in the PTLR

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure its functional requirements can still be satisfied using the PORV method or the depressurized and vented RCS condition.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of shutdown cooling (SDC); or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one HPSI pump, and all but one charging pump incapable of injection; and

b. Deactivating the SIT discharge isolation valves in their closed positions.

The Reference 3 analyses demonstrate that either one PORV or the RCS vent can maintain RCS pressure below limits when only one HPSI pump and one charging pump are actuated. Thus, the LCO allows only one HPSI pump and one charging pump OPERABLE during the LTOP MODES. Since neither the PORV nor the RCS vent can handle the pressure transient produced from accumulator injection, when RCS temperature is low, the LCO also requires the SITs isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated SITs must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of SIT discharge is

(continued)

Replace with
"the value in the PTLR"

BASES

APPLICABLE SAFETY ANALYSES

Heat Input Type Transients (continued)

over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([285]°F and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at [285]°F and below. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years of operation.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of one HPSI pump and one charging pump OPERABLE and SI actuation enabled for these pumps.

PORV Performance

Replace with
the value in the PTLR

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below [450] psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of one HPSI pump and one charging pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing setpoints, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the Reference 1 limits will be met.

The PORV setpoints will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement caused by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV represents the worst case, single active failure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of [1.3] square inches is capable of mitigating the limiting allowed LTOP overpressure transient. In that event, this size vent maintains RCS pressure less than the minimum RCS pressure on the P/T limit curve.

The RCS vent size will also be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

LTOP System satisfies Criterion 2 of the NRC Policy Statement.

LCO

This LCO is required to ensure that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires only one HPSI pump and one charging pump capable of injecting into the RCS and the SITs isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The elements of the LCO that provide overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs; or

b. The depressurized RCS and an RCS vent.

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set at [450] psig or less and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.

Replace with the value in the PTLR

(continued)

BASES

LCO (continued)

An RCS vent is OPERABLE when open with an area ≥ [1.3] square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

Replace with

as in the PTLR

Replace with:

the temperature found in the PTLR

This LCO is applicable in MODE 4 when the temperature of any RCS cold leg is ≤ [285]°F, in MODE 5, and in MODE 6 when the reactor vessel ~~head is on~~. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above [285]°F and below. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above [285]°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that SIT isolation is only required when the SIT pressure is greater than or equal to the RCS pressure for the existing temperature, as allowed by the P/T limit curves provided in the PTLR. This Note permits the SIT discharge valve surveillance performed only under these pressure and temperature conditions.

ACTIONS

A.1 and B.1

With two or more HPSI pumps capable of injecting into the RCS, overpressurization is possible.

The immediate Completion Time to initiate actions to restore restricted coolant input capability to the RCS reflects the importance of maintaining overpressure protection of the RCS.

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

C.1, D.1, and D.2

An unisolated SIT requires isolation within 1 hour. This is only required when the SIT pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR.

If isolation is needed and cannot be accomplished within 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed within 12 hours. By increasing the RCS temperature to > [175]°F, a SIT pressure of [600] psig cannot exceed the LTOP limits if the tanks are fully injected. Depressurizing the SIT below the LTOP limit stated in the PTLR also protects against such an event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

Replace with: as found in the PTLR

E.1

In MODE 4 when any RCS cold leg temperature is ≤ [285]°F, with one PORV inoperable, two PORVs must be restored to OPERABLE status within a Completion Time of 7 days. Two valves are required to meet the LCO requirement and to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time is based on the facts that only one PORV is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

(continued)

BASES

ACTIONS (continued)

F.1

The consequences of operational events that will overpressure the RCS are more severe at lower temperature (Ref. 6). Thus, one required PORV inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The 24 hour Completion Time to restore two PORVs OPERABLE in MODE 5 or in MODE 6 when the vessel head is on is a reasonable amount of time to investigate and repair several types of PORV failures without exposure to a lengthy period with only one PORV OPERABLE to protect against overpressure events.

G.1

If two required PORVs are inoperable, or if a Required Action and the associated Completion Time of Condition A, B, D, E, or F are not met, or if the LTOP System is inoperable for any reason other than Condition A through Condition F, the RCS must be depressurized and a vent established within 8 hours. The vent must be sized at least [1.3] square inches to ensure the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action protects the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time of 8 hours to depressurize and vent the RCS is based on the time required to place the plant in this condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, only one HPSI pump and all but [one] charging pump are verified OPERABLE with the other pumps locked out with power removed and the SIT discharge incapable of injecting into the RCS. The [HPI] pump[s] and charging pump[s] are rendered incapable

(continued)

SURVEILLANCE REQUIREMENTS

<u>SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3</u> (continued)

of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in [pull to lock] and at least one valve in the discharge flow path being closed.

The 12 hour interval considers operating practice to regularly assess potential degradation and to verify operation within the safety analysis.

<u>SR 3.4.12.4</u>

SR 3.4.12.4 requires verifying that the RCS vent is open ≥ [1.3] square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is unlocked open; or

b. Once every 31 days for a valve that is locked open.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance need only be performed if the vent is being used to satisfy the requirements of this LCO. The Frequencies consider operating experience with mispositioning of unlocked and locked vent valves, respectively.

<u>SR 3.4.12.5</u>

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve can be remotely verified open in the main control room.

The block valve is a remotely controlled, motor operated valve. The power to the valve motor operator is not required to be removed, and the manual actuator is not required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (continued)

locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure event.

The 72 hour Frequency considers operating experience with accidental movement of valves having remote control and position indication capabilities available where easily monitored. These considerations include the administrative controls over main control room access and equipment control.

SR 3.4.12.6

Performance of a CHANNEL FUNCTIONAL TEST is required every 31 days to verify and, as necessary, adjust the PORV open setpoints. The CHANNEL FUNCTIONAL TEST will verify on a monthly basis that the PORV lift setpoints are within the LCO limit. PORV actuation could depressurize the RCS and is not required. The 31 day Frequency considers experience with equipment reliability.

Replace with: the value in the PTLR

A Note has been added indicating this SR is required to be performed [12] hours after decreasing RCS cold leg temperature to ≤ [285]°F. The test cannot be performed until the RCS is in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every [18] months to adjust the whole channel so that it responds and the valve opens within the required LTOP range and with accuracy to known input.

The [18] month Frequency considers operating experience with equipment reliability and matches the typical refueling outage schedule.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. FSAR, Section [15].
4. 10 CFR 50.46.
5. 10 CFR 50, Appendix K.
6. Generic Letter 90-06.

5.6 Reporting Requirements

5.6.6

Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

- NOTE -

Reviewer's Note: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.
3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.

5.6 Reporting Requirements

5.6.6 RCS PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2σ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.

5.6.7

EDG Failure Report

[If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.]

5.6.8

PAM Report

When a report is required by Condition B or G of LCO 3.3.[17], "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9

Tendon Surveillance Report

[Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC

APPENDIX D

(TOTAL PAGES: 60)

METHODOLOGY TO CALCULATE RCS PRESSURE TRANSIENT DURING RCP START

1.0 PURPOSE

The purpose of this calculation is to provide analytical support for a study concerning RCS overpressure protection during low temperature modes of operation.

The calculation models the pressure transients which result during solid water RCS conditions when a RCP is started while the steam generator secondary inventories are at a higher temperature than the reactor vessel inventory. The analysis considers:

1. an RCS without liquid relief capability; and

2. systems with liquid relief valve capabilities with a valve set pressure of 465 psia (365 psia for Arkansas).

Beyond the first few time increments, calculations are performed via a computer code which was developed to model the subject transient. The format of the code parallels the herein contained analysis. Code listing is included in Appendix C, code input/output for plant specific analyses are found in Appendix D. Nomenclature is listed in Appendix A, while the Computer Code Certificate is found as Appendix B.

2.0 SCOPE

This calculation verifies the computer code "OVERP" using the Northeast Utilities (NEU) Millstone Unit No. 2 system parameters - as code input for conditions in which the RCS is solid water. Similar results are obtained for other plant systems by substituting the appropriate system parameters as code input. Results applicable to Arkansas Power and Light (AP&L) ANO Unit No. 2, Baltimore Gas and Electric (BG&E) Calvert Cliffs Units No. 1 and 2, and Omaha Public Power District (OPPD) Fort Calhoun Units No. 1 and2 are included in Section 9.5.

4.0 METHOD OF ANALYSIS

Subsequent to calculating the initial RCS mass and specific volume, a time dependent technique is used to model the primary coolant temperature throughout the RCS resulting when a single RCP is operated. A five node system is used to model the RCS. These nodes are:

a) operating RCP loop steam generator;
b) non-operating RCS loop steam generator;
c) reactor vessel annulus region;
d) reactor core; and
e) reactor vessel upper plenum.

4.0 Method of Analysis - continued.....

Properties at each of the above nodes are determined by appropriate energy balances. RCS pressure is computed as a function of total system energy content and specific volume.

When the RCS is a closed (no mass flux) system the specific volume remains constant.

When relief valves discharge mass from the system the computed RCS pressure after each time increment accounts for the mass release and energy convection. Liquid relief capacities are dependent on system pressure and temperature.

5.0 ASSUMPTIONS

1. All energy transferred from the secondary to primary coolant is absorbed totally by the primary coolant.

2. The steam generator heat transfer rate is a function of the average bulk ΔT; i.e., average bulk secondary temperature minus the average bulk primary temperature. This assumption treats each steam generator as a single node, resulting in a more conservative model than a multi-node log mean temperature difference analysis.

3. The overall heat transfer coefficient for each steam generator is invarient with time. Each coefficient is based on the initial flow through the respective steam generator.

4. Steam generator secondary water and metal masses which contribute as heat sources are assumed limited to the masses inside (and including) the SG downcomer.

5. Initially, the primary coolant "outside" of the shutdown cooling (SDC) nozzles (i.e. in the hot legs, cold legs, RCPs, SGs) is considered to be at a uniform temperature and equal to SG secondary temperature. The pressurizer and surge line are assumed saturated at RCS pressure.

6. Initially, primary coolant "inside" the SDC nozzles is considered at a uniform temperature (that of SDC, at maximum).

7. Upon starting, the RCP is conservatively assumed to attain full speed instantaneously. In fact, the time to reach full speed is approximately 10 seconds.

5.0 Assumptions - continued.....

8. At initiation of the transient, the SDC system is conservatively assumed to be isolated. Therefore, no heat is removed from the RCS by the SDC system. For additional conservatism, SDC system volumes are not considered.

9. Letdown flow paths are conservatively assumed isolated to allow for no mass release other than through relief valves.

10. Additional heat sources contributing to pressurization during the transient are:

    a. pressurizer heaters at full power (1500 KW for Millstone)

    b. 1% decay heat (25.6 MW for Millstone)

    c. RCP heat (5.0 MW for Millstone)

11. RCS boundaries are assumed rigid; i.e. no expansion at higher pressures and temperatures. Hence, specific volumes calculated in each step are low, thereby resulting in a conservative upper bound pressure over the duration of the transients.

12. Metal masses which bound the primary coolant are conservatively neglected as heat sinks.

13. Secondary is open to atmosphere. SG temperature is assumed to initially correspond to the saturated temperature of the developed hydraulic head at the tube sheet; thus T_{sec} = 220°F.

9.0 Body of Calculation

Shown in Figure 2 is a flow diagram of the pressure transient calculation. First, the initial conditions and constants must be calculated. Next, a single reactor coolant pump (RCP) is assumed to start instantaneously. After the first time increment (Δt) average property values at each of the system nodes are recalculated to compensate for the displaced volume elements. Heat transferred in the steam generators (SG1 and SG2) is calculated. The total system energy content is updated to include pressurizer heater, decay heat and RCP heat inputs in addition to the heat input from SG1 and SG2. Finally, the RCS pressure is computed as a function of the average RCS specific volume and energy content. The time increment concludes with a check for convergence of the primary and secondary temperatures

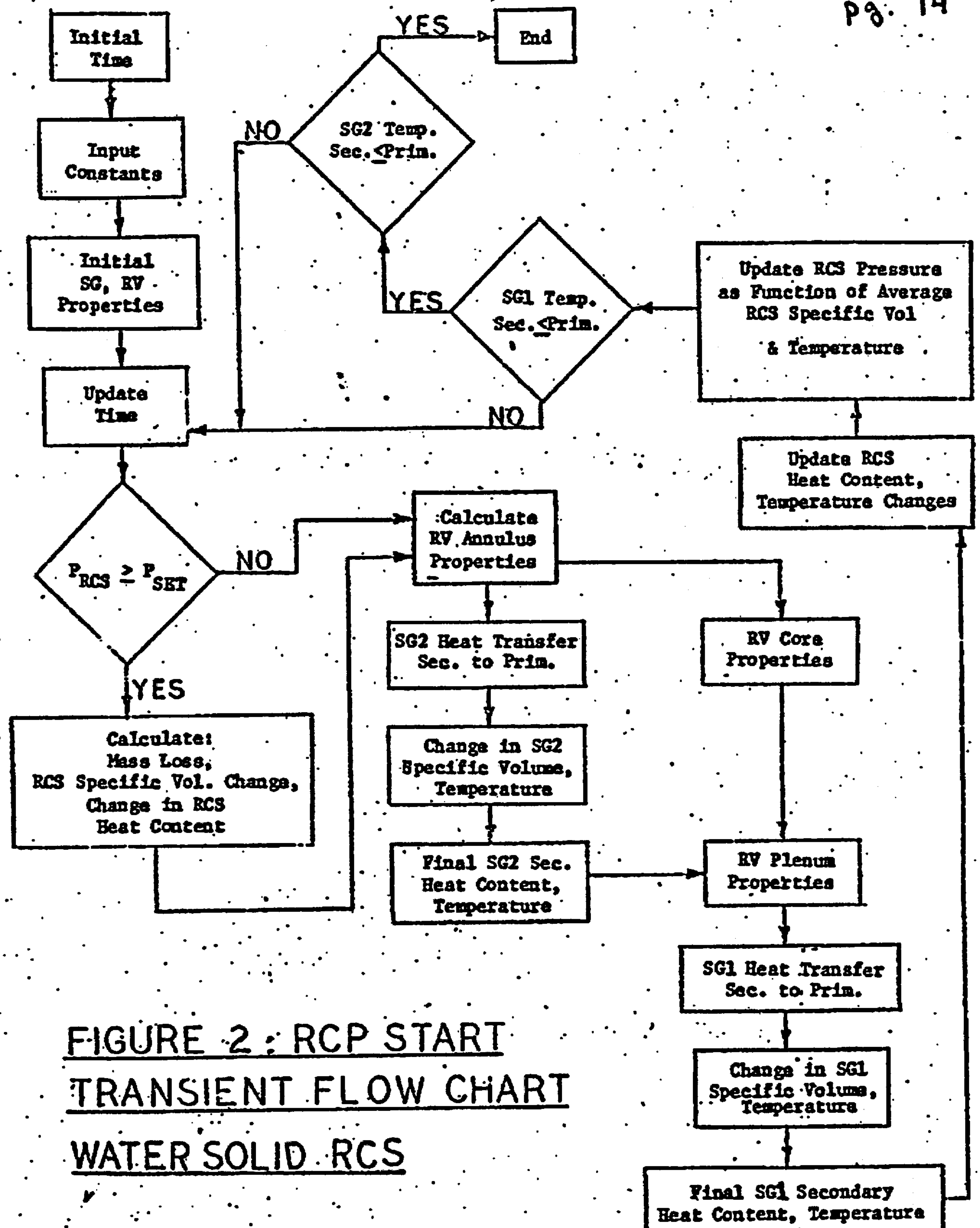


FIGURE 2 : RCP START TRANSIENT FLOW CHART WATER SOLID RCS

in each steam generator. The calculation is terminated when steam generator temperatures reach equilibrium.

At the beginning of each time increment the RCS pressure is compared to the relief valve set pressure (if relief valves are in system design). If RCS pressure is below set pressure the analysis is as described above. If the set pressure has been exceeded, the analysis compensates for the mass released and energy removed from the system by the valve(s) during the time increment.

9.2 ASSUMPTIONS

The following assumptions concerning flow paths and fluid properties are in addition to those stated in Section 5.0. Reference should be made to Figure 4 of the following page.

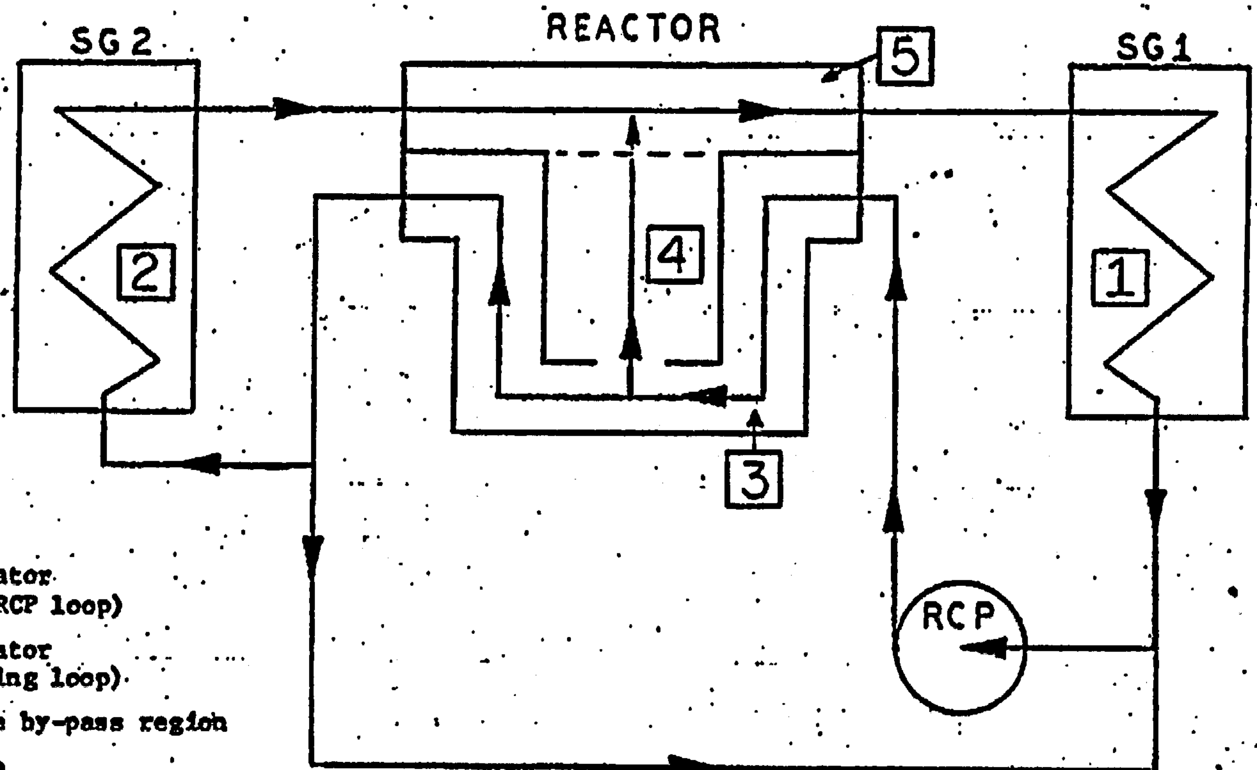
1) Upon starting, the RCPs attain full speed instantaneously. (The actual time to reach full speed is ≈ 10 seconds).

2) At the beginning of each time increment, SG1 active tube inlet properties assume the values at the reactor vessel hot leg exit from the previous time increment.

3) The entering volume element to the reactor vessel annulus region assumes the properties of the SG1 exit values from the current time increment.

4) The entering volume element to the reactor vessel core inlet assume the properties of the RV annulus after mixing has been considered.

FIGURE 4 : MODEL NODES FOR RCP START TRANSIENT



Node	Description
1	steam generator (operating RCP loop)
2	steam generator (non-operating loop)
3	annulus core by-pass region
4	reactor core
5	upper plenum

NOTE:

1. Flow split values are indicated in Figure 3

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5) The entering volume element to the SG2 inlet assume the properties of the reactor vessel annulus after mixing.

6) Reactor vessel upper plenum inlet properties assume the mixed value from the SG2 and RV core regions of the current time increment

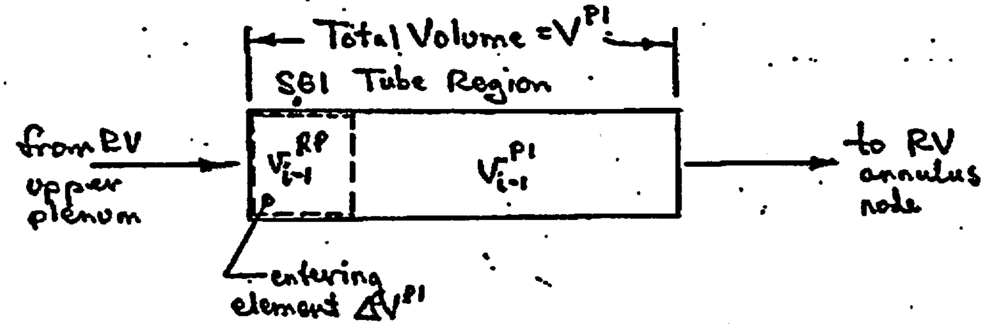
9.3 DERIVATION OF EQUATIONS

The equations used in this analysis are derived in the order in which each is applied throughout the course of the calculation. Reference should be made to the flow chart of Figure 2, the flow splits of Figure 3, and the nodal diagram of Figure 4. Time dependent variables are indicated by the subscripts "i", "i+1", and "i-1". Initial conditions are indicated by a zero subscript designation. For nomenclature definition, reference should be made to Appendix C.

9.3.1 SG1 NODE

The calculation first initializes RCS properties as in Section 9.1. A volume element of primary coolant from the reactor vessel upper plenum region then displaces an element in SG1. The primary coolant contained in the SG1 tubes is assumed to instantaneously mix with the entering volume element, resulting in an average temperature change. The "mixed" specific volume and temperature are found as follows:

AVERAGE SPECIFIC VOLUME, v_i^{P1}:



$$\text{Average Specific Volume, } v_i^{P1} = \frac{\text{Total Tube Volume}}{\text{Total Mass}}$$

$$\text{Mass of entering element} = \frac{\Delta V^{P1}}{v_{i-1}^{RP}}$$

Mass of Resident Fluid $= \dfrac{V^{PI} - \Delta V^{PI}}{v_{i-1}^{PI}}$

Therefore, $v_i^{PI} = \dfrac{V^{PI}}{\dfrac{\Delta V^{PI}}{v_{i-1}^{RP}} + \dfrac{V^{PI} - \Delta V^{PI}}{v_{i-1}^{PI}}}$

Which can be expressed as:

$$v_i^{PI} = \frac{V^{PI} \cdot v_{i-1}^{RP}\; v_{i-1}^{PI}}{\Delta V^{PI} v_{i-1}^{PI} + (V^{PI} - \Delta V^{PI}) v_{i-1}^{RP}}.$$

AVERAGE TEMPERATURE, $T_{i_{mix}}^{PI}$:

T^{PI} may be found from Reference 5 steam tables using the RCS pressure at the previous time increment and the specific volume calculated above. Thus, the temperature resulting from mixing is

$$T_{i_{mix}}^{PI} = f(P_{i-1}^{RCS}, v_i^{PI})$$

From $\dot{q} = UA\Delta T$, the heat transferred from the secondary side to the primary is calculated based on:

$$\Delta T = T_{i-1}^{SI} - T_{i_{mix}}^{PI} = \Delta T_i^{S/P}$$

Where $T_{i_{mix}}^{PI}$ results from instantaneous mixing of incoming volume element.

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Over a time increment, Δt, the heat transferred is $\int \frac{dq}{dt} dt = \int U A \Delta T^{s/p} dt$ or $q = U A \Delta T (\Delta t)$ (eq. 1)

On the other hand, any heat transferred from the secondary should result in an increase of primary fluid temperature. The equation which applies for this case is:

$$\Delta q^{p} = m_{w}^{p} C_{p} \Delta T^{p} \quad \text{(eq. 2)}$$

where $\Delta T^{p} = T_{i-1}^{p1} - T_{imix}^{p1}$.

It is noted, however, that the heat transferred from the secondary to the primary (eq. 1) must be equal to that heat absorbed by the primary (from eq. 2) over the time increment and also equal to the heat loss from the secondary. In addition to (eq. 2), the following also applies

$$\Delta q^{s} = (m_{w}^{s} C_{p} + m_{m}^{s} C) \Delta T^{s} \quad \text{(eq. 3)}$$

where $\Delta T^{s} = T_{i-1}^{s1} - T_{i}^{s1}$.

Since the three noted equations must all balance, a unique average $\Delta T^{S/P}$ must exist over the time increment for which the increase in primary coolant heat content, $\delta(T_{i-1}^{P1} - T_{i,mix}^{P1})$, balances with the heat transferred from the secondary as found in eq.(1). The $q = \delta(\Delta T^{S/P})$ must also balance with $\delta(T_{i-1}^{S1} - T_i^{S1})$, found in eq.(3).

It is noted that T_i^{P1} results not only from a simple mixing of the displaced volume element in the tubes, but also from the heat absorbed simultaneously from the secondary side. Consequently, the T_i^{P1} initially calculated is not the final T_i^{P1} in the time step upon which heat transfer is based. The final T_i^{P1}, and also T_i^{S1}, which will determine the $\Delta T^{S/P}$ to balance eq.(1), eq(2), and eq(3) must be determined by an 'iterative' calculation. The iteration scheme is presented as follows:

STEAM GENERATOR HEAT TRANSFER ITERATION

The description of the following iteration will apply to both SG1 and SG2.

Heat transfer due to the original secondary to mixed primary temperature difference is

$$\Delta q_i^{SG1} = U^{SG1} A_s \left[T_{i-1}^{S1} - T_{i\,mix}^{P1} \right] \Delta t \qquad (\text{eq } 4a)$$

where $T_{i\,mix}^{P1}$ results from mixing only and T_{i-1}^{S1} is the original secondary temperature at the beginning of the time increment.

Input of eq(4a) heat into the primary coolant during the time increment would result in an increase of primary temperature and a decrease of secondary temperature as follows:

From $q = mc_p \Delta T$

$$q_{in}^{P} = \Delta q_i^{SG1} = m_i^{P1} c_p \left(T_i^{P1} - T_{i\,mix}^{P1}\right)$$

Solving for T_i^{P1}

$$T_i^{P1} = \frac{\Delta q_i^{SG1}}{m_i^{P1} \cdot c_p} + T_{i\,mix}^{P1} \qquad \text{(eq. 4b)}$$

Likewise, applying Δq_i^{SG1} to the secondary metal and liquid:

$$q_{out}^{S} = \Delta q_i^{SG1} = m_m^S c_m \left(T_{i-1}^{S1} - T_i^{S1}\right) + m_w^S c_p \left(T_{i-1}^{S1} - T_i^{S1}\right)$$

Rearranging and solving for T_i^{S1}

$$\Delta q_i^{SG1} = \left(m_m^S c_m + m_w^S c_p\right)\left(T_{i-1}^{S1} - T_i^{S1}\right)$$

let $c_{mw} = m_m^S c_m + m_w^S c_p$

$$T_i^{S1} = T_{i-1}^{S1} - \frac{\Delta q_i^{SG1}}{c_{mw}} \qquad \text{(eq. 4c)}$$

It follows that the temperature difference upon which eq. (4a) is based will change since T_i^{P1} and T_i^{S1} have been revised by eq (4b) and (4c). Consequently, an iteration develops which calculates

Δq_i^{SG1} as a function of T_i^{P1} and T_i^{S1} of the previous iteration. This iteration should continue until convergence of temperatures such that Δq_i^{SG1} does not change any further (or does not change appreciably). During these iterations, the $T_{i\,mix}^{P1}$ of eq.(4a) becomes the T_i^{P1} calculated in eq(4b); the $T_{i\,mix}^{P1}$ in equation (4b) remains the same after each iteration. Likewise, the T_{i-1}^{S1} remains the same after each iteration.

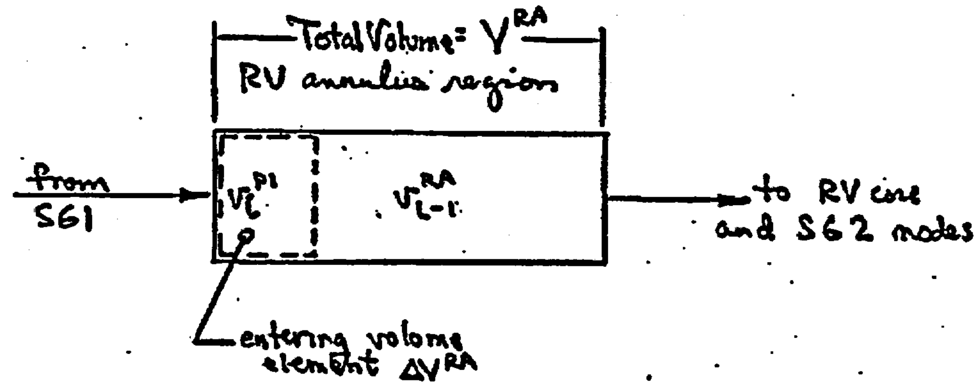
UPDATED SG PRIMARY SPECIFIC VOLUME, v^{P1}:

Since primary coolant temperature has changed as a result of heat transfer to the coolant during the time increment; in addition to mixing, the specific volume is updated using STEAM properties:

$$v_i^{P1} = f(P_{i-1}^{RCS}, T_i^{P1})$$

9.3.2 Reactor Vessel Annulus Node

During each time increment a fluid element enters the reactor vessel annulus region from the operating RCP loop. It is assumed that the fluid temperature is the same as the average mixed temperature exiting the SG1 tube region; thus, the specific volume is v_i^{P1}. The average specific volume in the RV annulus region is found as follows:



$$\text{Average specific volume, } v_i^{RA} = \frac{\text{Total Annulus Volume}}{\text{Total Mass}}$$

$$\text{Mass of entering element} = \frac{\Delta V^{RA}}{v_i^{P1}}$$

$$\text{Mass of resident fluid} = \frac{V^{RA} - \Delta V^{RA}}{v_{i-1}^{RA}}$$

Therefore, $$v_i^{RA} = \frac{V^{RA}}{\dfrac{\Delta V^{RA}}{v_i^{PI}} + \dfrac{V^{RA} - \Delta V^{RA}}{v_{i-1}^{RA}}}$$

Which can be expressed as:

$$v_i^{RA} = \frac{V^{RA}\ v_i^{PI}\ v_{i-1}^{RA}}{\Delta V^{RA}\ v_{i-1}^{RA} + (V^{RA} - \Delta V^{RA})\, v_i^{PI}}$$

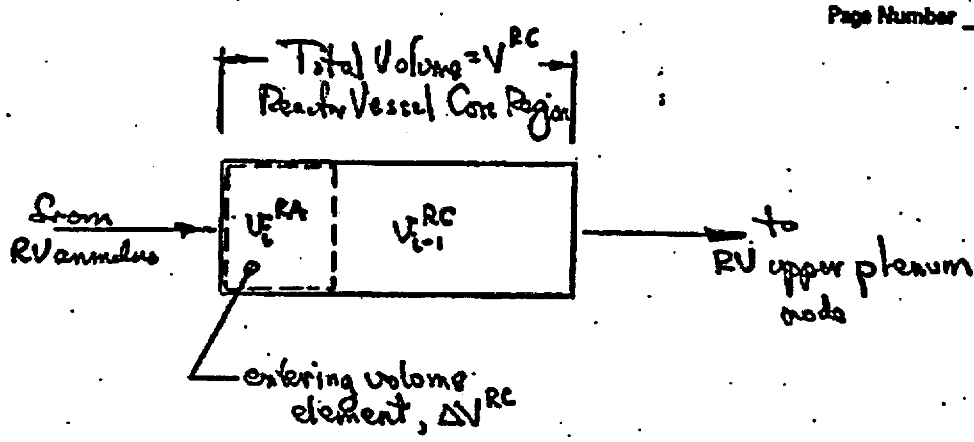
where $\Delta V^{RA} = \Delta V^{PI}$

This specific volume is used as the entering element specific volume at the SG2 and RV core nodes during the current time step.

9.3.3 REACTOR VESSEL CORE NODE

The entering volume element to the RV core region is assumed to have the same specific volume as the average RV annulus specific volume from the current time step, i.e. v_i^{RA}.

Thus, the average specific volume in the RV core region is found as follows:



Average Specific Volume, $v_i^{RC} = \dfrac{\text{Total Core Volume}}{\text{Total Mass}}$

Mass of entering element $= \dfrac{\Delta V^{RC}}{v_i^{RA}}$

Mass of resident fluid $= \dfrac{V^{RC} - \Delta V^{RC}}{v_{i-1}^{RC}}$

Therefore, $v_i^{RC} = \dfrac{V^{RC}}{\dfrac{\Delta V^{RC}}{v_i^{RA}} + \dfrac{V^{RC} - \Delta V^{RC}}{v_{i-1}^{RC}}}$

Which can be reduced to:

$$v_i^{RC} = \frac{V^{RC}\, v_i^{RA}\, v_{i-1}^{RC}}{\Delta V^{RC}\, v_{i-1}^{RC} + (V^{RC} - \Delta V^{RC})\, v_i^{RA}}$$

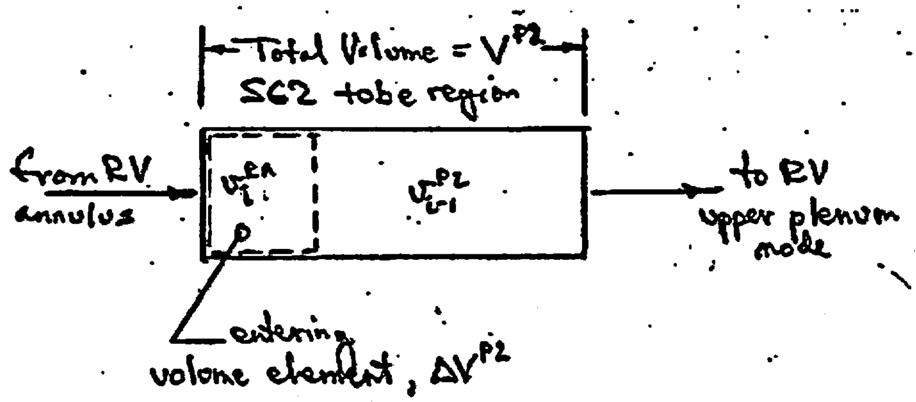
where $\Delta V^{RC} = \Delta V^{P1} - \Delta V^{P2}$

This specific volume is the value used entering the reactor vessel upper plenum node region during the current time step.

9.3.4 SG2 NODE

The entering volume element to the SG2 tube region assumes the same specific volume as the average RV annulus specific volume of the current time step; i.e. v_i^{RA}. As in SG.1, the entering volume element to SG2 instantaneously mixes with the remaining resident fluid in the region, resulting in an average temperature change. The "mixed" specific volume and temperature are as follows:

AVERAGE SPECIFIC VOLUME ; v_i^{P2}:



$$\text{Average Specific Volume}, v_i^{P2} = \frac{V^{P1}}{\text{Total Fluid Mass in tube Region}}$$

Mass of entering element $= \dfrac{\Delta V^{P2}}{v_i^{RA}}$

Mass of fluid resident in SG2 tube region $= \dfrac{V^{P2} - \Delta V^{P2}}{v_{i-1}^{P2}}$

Therefore, $v_i^{P2} = \dfrac{V^{P2}}{\dfrac{\Delta V^{P2}}{v_i^{RA}} + \dfrac{V^{P2} - \Delta V^{P2}}{v_{i-1}^{P2}}}$

Which can be expressed as:

$$v_i^{P2} = \frac{V^{P2}\, v_i^{RA}\, v_{i-1}^{P2}}{\Delta V^{P2}\, v_{i-1}^{P2} + (V^{P2} - \Delta V^{P2})\, v_i^{RA}}$$

This "mixed" specific volume is used to find the average temperature in the SG2 node region as follows.

AVERAGE TEMPERATURE ; $T_{i,mix}^{P2}$:

$T_{i,mix}^{P2}$ may be found from steam property tables using the RCS pressure at the previous time increment and the specific volume calculated above. Thus, the average temperature resulting from mixing is: $T_{i,mix}^{P2} = f(P_{i-1}^{RCS}, v_i^{P2})$.

SG2 HEAT TRANSFER ITERATION

Using the same method of iteration as for SG1 to find the secondary to primary temperature differential, increase in primary temperature and decrease in secondary temperature the following equations result:

$$\Delta q_i^{SG2} = U^{SG2} A_s \left[T_{i-1}^{S2} - T_{i\,mix}^{P2} \right] \Delta t \qquad \text{(eq. 5a)}$$

$$T_i^{P2} = \frac{\Delta q_i^{SG2}}{m_i^{P2} C_p} + T_{i\,mix}^{P2} \qquad \text{(eq. 5b)}$$

$$T_i^{S2} = T_{i-1}^{S2} - \frac{\Delta q_i^{SG2}}{C_{mw}} \qquad \text{(eq. 5c)}$$

where $C_{mw} = m_m^s C_m + m_w^s C_p$

During each iteration, the temperature difference upon which eq (5a) is based will change since T_i^{P2} and T_i^{S2} have been revised by eqs. (5b) and (5c). Consequently, Δq_i^{SG2} is calculated as a function of T_i^{P2} and T_i^{S2} of the previous iteration. The iteration should continue until convergence of temperatures of successive iterations such that Δq_{Bi}^{SG2} does not change any further (or does not change appreciably). During these iterations, the $T_{i_{mix}}^{P2}$ of eq. (5a) becomes the T_i^{P2} calculated in eq. (5b); the $T_{i_{min}}^{P2}$ in equation (5b) remains the same after each iteration. Likewise, the T_{i-1}^{S2} in eq (5c) remains the same after each iteration.

UPDATED SG2 PRIMARY SPECIFIC VOLUME, v_i^{P2}:

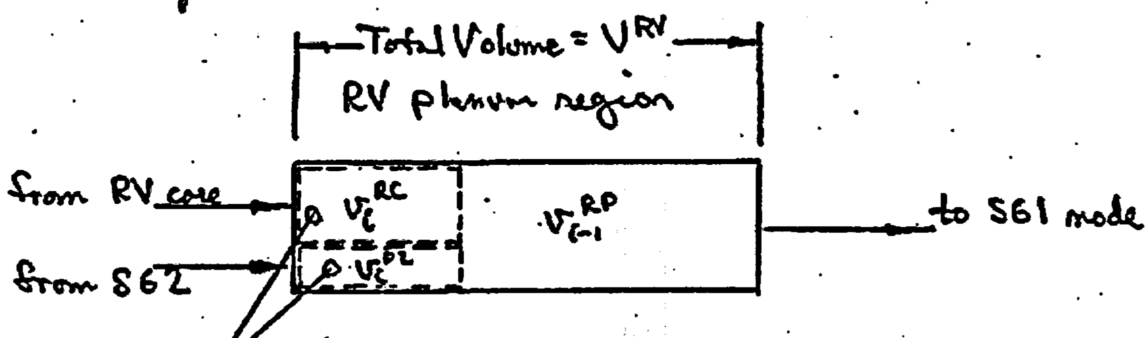
In addition to mixing, the primary coolant temperature has changed as a result of heat transfer to the coolant during the time increment. Therefore the specific volume is finally updated using steam properties:

$$v_i^{P2} = f\left(P_{i-1}^{RCS}, T_i^{P2}\right)$$

This specific volume is one of the values used entering the reactor vessel upper plenum region during the next time increment.

9.3.5 REACTOR VESSEL UPPER PLENUM NODE

Volume elements enter the RV upper plenum region by two flow paths: 1) from the SG2 tube region, and 2) from the RV core region. The volume elements assume the respective specific volumes from these regions as calculated in the previous sections. The average specific volume in the plenum region, v^{RV}, is calculated as follows:



entering volume elements: ΔV^{RP1} (from RV core) $= \Delta V^{RC} = \Delta V^{P1} - \Delta V^{P2}$

ΔV^{RP2} (from SG2) $= \Delta V^{P2}$

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Average Specific Volume, v^{RP} = $\dfrac{\text{Total Plenum Volume}}{\text{Total Mass}}$

Mass of entering element from RV core = $\dfrac{\Delta V^{RC}}{v_i^{RC}}$

Mass of entering element from 862 = $\dfrac{\Delta V^{P2}}{v_i^{P2}}$

Mass of RV plenum remaining = $\dfrac{V^{RP} - \Delta V^{RC} - \Delta V^{P2}}{v_{i-1}^{RP}}$

Therefore, $v_i^{RP} = \dfrac{V^{RP}}{\dfrac{\Delta V^{RC}}{v_i^{RC}} + \dfrac{\Delta V^{P2}}{v_i^{P2}} + \dfrac{V^{RP} - \Delta V^{RC} - \Delta V^{P2}}{v_{i-1}^{RP}}}$

Which can be expressed as:

$$v_i^{RP} = \frac{V^{RP}\, v_i^{RC}\, v_i^{P2}\, v_{i-1}^{RP}}{\Delta V^{RC} v_i^{P2} v_{i-1}^{RP} + \Delta V^{P2} v_i^{RC} v_{i-1}^{RP} + (V^{RP} - \Delta V^{RC} - \Delta V^{P2})(v_i^{RC}\, v_i^{P2})}$$

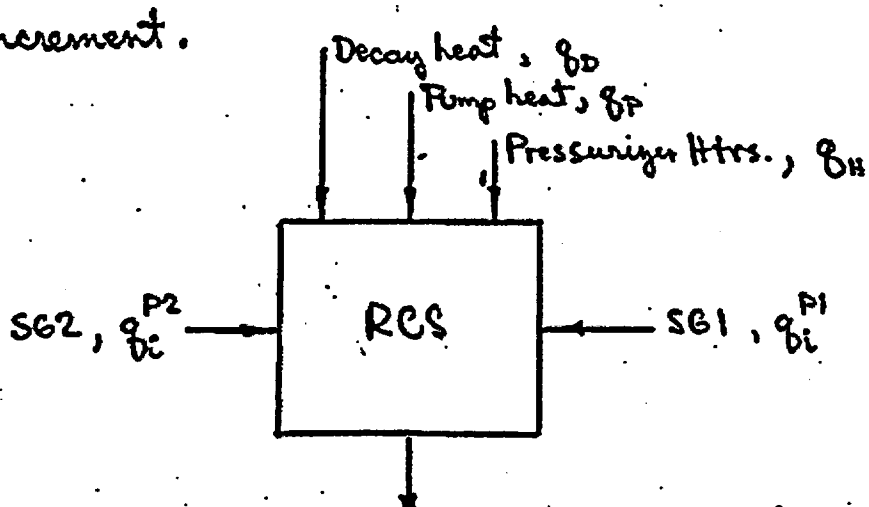
Since $\Delta V^{RP} = (\Delta V^{RC} + \Delta V^{P2}) = (\Delta V^{P1} - \Delta V^{P2}) + \Delta V^{P2} = \Delta V^{P1}$

$$v_i^{RP} = \frac{V^{RP}\, v_i^{RC}\, v_i^{P2}\, v_{i-1}^{RP}}{\Delta V^{RC} v_i^{P2} v_{i-1}^{RP} + \Delta V^{P2} v_i^{RC} v_{i-1}^{RP} + (V^{RP} - \Delta V^{P1})(v_i^{RC} v_i^{P2})}$$

9.3.6 RCS ENTHALPY CHANGE DURING TIME INCREMENT

After the specific volume at each RCS node is updated, and the heat transferred in SG1 and SG2 calculated, the total RCS energy content is updated.

The following sketch illustrates the various energy inputs considered for each time increment.



Relief Valve Discharge, $m_{i-1}^{out} = f(P_{i-1}^{RCS}, \Delta t)$

$$q_i^{out} = m_{i-1}^{out} h_{pzr} \Delta t$$

The time increment change in enthalpy is:

$$\Delta h_i^{RCS} = \frac{\Delta q_i^{RCS} + m_{i-1}^{out} h_{i-1}^{RCS} \Delta t}{m_i^{RCS}}$$

where $\Delta q_i^{RCS} = q_i^{P1} + q_i^{P2} + q_D + q_P + q_H - q_i^{out}$

$$q_i^{out} = m_{i-1}^{out} \, h^{PZR} \, \Delta t$$

$$m_i^{RCS} = m_{i-1}^{RCS} - m_{i-1}^{out} \Delta t$$

Relief valve discharge, m_{i-1}^{out}, is discussed in a later section.

RCS total enthalpy after each time increment becomes,

$$h_i^{RCS} = h_{i-1}^{RCS} + \Delta h_i^{RCS}$$

9.3.7 RCS PRESSURE UPDATE

The RCS enthalpy is used to calculated the average RCS temperature:

From $\Delta q = mC\Delta T$ and $\Delta h = \Delta q / m$

$$\Delta h_i^{RCS} = C_p \left(T_i^{RCS} - T_{i-1}^{RCS} \right)$$

Solving for T_i^{RCS}

$$T_i^{RCS} = \frac{\Delta h_i^{RCS}}{C_p} + T_{i-1}^{RCS}.$$

The RCS pressure after each time increment is found as a function of temperature and specific volume using Reference 5.

i.e. $P_i^{RCS} = f(T_i^{RCS}, v_i^{RCS})$.

When relief valve discharges are involved, the system specific volume will change according to the release of fluid.

Thus, $v_i^{RCS} = \dfrac{V^{RCS}}{m_{i-1}^{RCS} - \Delta m_{i-1}^{out}}$

where $\Delta m_{i-1}^{out} = \dot{m}_{i-1}^{out} \Delta t$

After RCS pressure is found the calculation proceeds to the next time increment, beginning with the SG1 node described in Section 9.3.1. The pressure transient will terminate when the RCS reaches thermal equilibrium. If decay heat, pressurizer heater input, or pump heat is considered negligible or non-existent, thermal equilibrium between SG primary and secondary inventories will terminate the transient.

9.3.8 RELIEF VALVE DISCHARGES

Mass releases are modeled as either a mechanical spring-loaded relief valve (SRV) or an electromatic power operated relief valve (PORV). The PORV opens instantaneously once RCS pressure exceeds the valve set pressure. The SRV is assumed to gradually open as system pressure exceeds the set pressure; the valve is assumed full open once pressure reaches 10% accumulation. Specific discussion for each valve model follows.

Pressure at the valve inlet is assumed as PZR. pressure. The discharges of the PORV and SRV are represented as a Bernoulli orifice discharge:

$$m^{out} = 0.525\, d^2\, C \sqrt{(\Delta P)\rho} \quad \frac{\#_m}{sec} \quad \text{Eqn (6)}.$$

where $d \equiv$ valve orifice equivalent diameter; inch.

$C \equiv$ valve flow coefficient. A value of 0.60 is conservatively assumed for liquid flow.

$\Delta P = P^{RCS} - P_B$; $P^{RCS} \equiv$ indicated pressurizer pressure (psig).

$P_B \equiv$ backpressure = Quench tank design pressure plus piping ΔP ≤ 100 psig.

P_B is conservatively assumed constant.

$\rho \equiv$ liquid density. Assumed as the saturated density in the pressurizer at the initiation of the transient (i.e. 52.926 $\#_m/ft^3$ @ 300psia)

PORV MODEL:

The PORV capacity is modeled as a parabolic function of the pressure differential across the valve. Thus, in the "i^{th}" time increment:

$$m_i^{out} = m^{Rated} \sqrt{\frac{P_{i-1}^{RCS} - P_B}{P_{set} - P_B}}$$

where m^{Rated} is found from Eqn (6) when P^{RCS} is equal to the valve set pressure, P_{set} (used as a reference pressure for the rated capacity). Thus, for P_{set} = 450 psig:

$$m_{rated} = 0.525\,d^2\,(0.60)\sqrt{(450-100)(52.926)}$$

$$m_{rated} = 42.873\,d^2 \qquad \text{Eqn (7).}$$

The orifice areas and diameters* as related by the valve vendor (Dresser Valve Co.) are as follows:

Plant	Orifice Area (in²)	Bore Dia (in)	Seat Dia (in)	Comments
OPPD	0.940	1 3/32	1 5/8	non bellows
NEU	1.354	1 5/16	1 5/16	bellows, full bore
BG&E	1.289	1 9/32	1 5/16	bellows
CPCo.	1.485	1 3/8	1 5/8	non-bellows
FP&L	1.354	1 5/16	1 5/16	bellows, full bore

* Bore diameter is actual flow diameter, seat diameter is maximum possible.

Using the NEU valve as an example, the rated valve discharge flow becomes:

$$m_{rated}^{NEU} = 42873(1\,{}^5/_{16})^2 = 73.86\ \#m/sec$$

The valve flowrate at the "i^{th}" time increment may then be expressed as a function of the rated flow rate and the RCS pressure:

$$m_i^{out} = m^{rated}\sqrt{\frac{P_{i-1}^{RCS} - P_B}{P_{set} - P_B}}$$

Pursuing the NEU example,

$$m_i^{out} = 73.86\sqrt{\frac{P_{i-1}^{RCS} - 100}{450 - 100}}$$

$$m_i^{out} = 3.95\sqrt{P_{i-1}^{RCS} - 100}\quad \#m/sec$$

where P_{i-1}^{RCS} is expressed in PSIG units.

Equations for BG&E and OPPD are similar except that the following rated capacities apply:

$m_{rated}^{BG\&E} = 70.3\ \#m/sec$; $m_{rated}^{OPPD} = 51.3\ \#m/sec$

due to differences in orifice diameters.

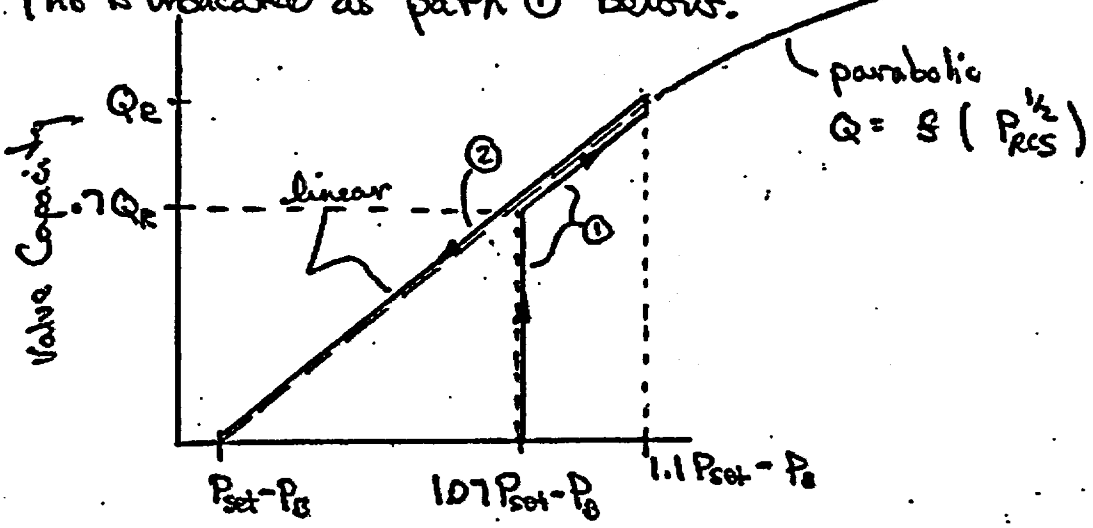
SRV MODEL

Characteristically, opening and closing behavior for spring relief valves will differ for steam and water applications. Typically, steam valves will attain full lift at 3% accumulation above setpressure. Steam valves will close within 5% blowdown below setpressure. On the other hand, liquid valve full lift accumulation is 10% and blowdown is ≈ 10%. For both steam and water the rated valve capacity is typically given at the full lift accumulation.

In order to accurately estimate the liquid valve behavior between set pressure and full lift CE requested that Bechtel pursue the topic with a valve vendor. The results of this inquiry were related from Bechtel to CE.

The valve vendor (Crosby Valve Co.) will not guarantee that the spring valve will begin to open until 7% accumulation. At this point the valve is guaranteed to be lifted; however, its opening behavior to full lift at 10% accumulation is unknown. Based on this information the following valve model was developed:

At 7% accumulation the valve "pops" to a position where the capacity is defined by the coordinates located on a linear line between zero and the rated capacity at 10% accumulation. This is indicated as path ① below.



Once accumulation of 10% is attained the valve is fully open and discharge is assumed to proceed parabolically.

The valve is assumed to close at the set pressure. This will conservatively minimize the quantity of mass discharge, rather than continuing relief until 10% below set pressure. Flow path ② is followed for decreasing pressures

The linear portion of the capacity curve is characterized by the general equation

$$y = mx + b$$

where $m = \frac{\Delta Y}{\Delta X} = \frac{Q_R - 0}{(1.1 P_{set} - P_B) - (P_{set} - P_B)} = 10 \frac{Q_R}{P_{set}}$

$$b = y - mx = -10 \frac{Q_R}{P_{set}} (P_{set} - P_B)$$

Then for $Q = f(P_{RCS})$

$$Q = 10 \frac{Q_R}{P_{set}} (P_{RCS} - P_B) - 10 Q_R \left(\frac{P_{set} - P_B}{P_{set}} \right)$$

or, $Q = 10 Q_R \left[\frac{P_{RCS}}{P_{set}} - 1 \right]$ eq.(8)

This flow rate is expressed as a volumetric quantity. For mass flow rate the appropriate conversion factor must be applied and the volume term multiplied by the liquid density at the valve inlet.

The parabolic portion of the capacity curve is characterized in the same manner as for the PORV. That is,

$$Q = Q_{rated}\sqrt{\frac{P_{RCS} - P_{Back}}{1.1\,P_{set} - P_{Back}}} \qquad \text{eqn(9)}$$

The reference ΔP for Q_{rated} is $(1.1\,P_{set} - P_{Back})$. This is 10% accumulation minus the back pressure.

For the case of Arkansas Power and Light, the required valve flow rate at 10% accumulation is 1600 gpm (defined by an inadvertent safety injection). Set pressure is 365 psia and maximum back pressure is 150 psia.

Maximum temperature at the valve inlet is assumed the temperature in the pressurizer at initiation of the pressure transient. Maximum PZR pressure is 300 psia; saturated temperature is 417 °F, The density at 401.5 psia (10% accumulation) and 417°F is 52.97 $\frac{lb_m}{ft^3}$.

T_{sat} at 401.5 psia is ~ 445 °F.

Thus, at the valve inlet the liquid is at least 445 - 417 = 28°F subcooled.

Thus, flashing in the valve throat is not assumed and equation (9) is used.

Q_{rated} in eqns (8) and (9) is 1600 gpm. Expressed as a mass discharge this becomes,

$$\dot{m}_{rated} = 1600 \frac{\cancel{gal}}{\cancel{min}} \times \frac{1 \cancel{min}}{60 sec} \times 0.1337 \frac{\cancel{ft^3}}{\cancel{gal}} \times 52.97 \frac{lb_m}{\cancel{ft^3}}$$

$$\dot{m}_{rated} = 188.9 \ lb_m/sec.$$

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9.3.9 SUMMARY OF EQUATIONS

The equations presented in the previous sections are summarized below.

1. $$v_i^{P1} = \frac{V^{P1}\, v_{i-1}^{RP}\, v_{i-1}^{P1}}{\Delta V^{P1}\, v_{i-1}^{P1} + (V^{P1} - \Delta V^{P1})\, v_{i-1}^{RP}}$$

2. $$T_{i_{mix}}^{P1} = f(P_{i-1}^{RCS},\ v_i^{P1})$$

3. $$\Delta q_{bi}^{SG1} = U^{SG1} A_S \left[T_{i-1}^{S1} - T_{i_{mix}}^{P1} \right] \frac{\Delta t}{3600}$$

4. $$T_i^{P1} = \frac{\Delta q_{bi}^{SG1}}{m_i^{P1} C_p} + T_{i_{mix}}^{P1} \ ; \quad m_i^{P1} = \frac{V^{P1}}{v_i^{P1}}$$

ITERATION

5. $$T_i^{S1} = T_{i-1}^{S1} - \frac{\Delta q_{bi}^{SG1}}{C_{mw}} \ ; \quad C_{mw} = m_m^S C_m + m_w^S C_p$$

6. $$v_i^{P1} = f(P_{i-1}^{RCS},\ T_i^{P1})$$

7. $$v_i^{RA} = \frac{V^{RA}\, v_i^{P1}\, v_{i-1}^{RA}}{\Delta V^{P1}\, v_{i-1}^{RA} + (V^{RA} - \Delta V^{P1})\, v_i^{P1}}$$

8. $$v_i^{RC} = \frac{V^{RC}\, v_i^{RA}\, v_{i-1}^{RC}}{\Delta V^{RC}\, v_{i-1}^{RC} + (V^{RC} - \Delta V^{P1} + \Delta V^{P2})\, v_i^{RA}}$$

9. $$v_i^{P2} = \frac{V^{P2}\, v_i^{RA}\, v_{i-1}^{P2}}{\Delta V^{P2}\, v_{i-1}^{P2} + (V^{P2} - \Delta V^{P2})\, v_i^{RA}}$$

10. $$T_{i_{mix}}^{P2} = f(P_{i-1}^{RCS},\ v_i^{P2})$$

11. $\Delta q_{bi}^{SG2} = U^{SG2} A_s \left[T_{i-1}^{S2} - T_{i_{mix}}^{PZ} \right] \frac{\Delta t}{3600}$

12. $T_i^{PZ} = \frac{\Delta q_{bi}^{SG2}}{m_i^{PZ} C_p} + T_{i_{mix}}^{PZ}$; $m_i^{PZ} = \frac{V^{PZ}}{v_i^{PZ}}$ ITERATION

13. $T_i^{S2} = T_{i-1}^{S2} - \frac{\Delta q_{bi}^{SG2}}{C_{mw}}$

14. $v_i^{PZ} = f\left(P_{i-1}^{RCS}, T_i^{PZ}\right)$

15. $v_i^{RP} = \frac{V^{RP}\, v_i^{RC}\, v_i^{PZ}\, v_{i-1}^{RP}}{\Delta V^{RC} v_i^{PZ} v_{i-1}^{RP} + \Delta V^{PZ} v_i^{RC} v_{i-1}^{RP} + \left(V^{RP} - \Delta V^{P1}\right) v_i^{RC} v_i^{PZ}}$

16. $m_{i-1}^{out} = f\left(P_{i-1}^{RCS}, \text{Orifice Area}\right)$ see Section 9.3.8

17. $\Delta q_{bi}^{out} = m_{i-1}^{out}\, h_{PZR}\, \Delta t$

18. $\Delta q_{bi}^{RCS} = \Delta q_{bi}^{SG1} + \Delta q_{bi}^{SG2} + \Delta q_{bD} + \Delta q_{bH} - \Delta q_{bi}^{out}$

19. $m_i^{RCS} = m_{i-1}^{RCS} - m_{i-1}^{out}\, \Delta t$

20. $\Delta h_i^{RCS} = \frac{\Delta q_{bi}^{RCS} + m_{i-1}^{out}\, h_{i-1}^{RCS}\, \Delta t}{m_i^{RCS}}$

21. $T_i^{RCS} = \frac{\Delta h_i^{RCS}}{C_p} + T_{i-1}^{RCS}$

22. $v_i^{RCS} = \frac{V^{RCS}}{m_{i-1}^{RCS} - \Delta m_{i-1}^{OUT}}$

23. $P_i^{RCS} = f\left(T_i^{RCS}, v_i^{RCS}\right)$

9.4 SAMPLE CALCULATIONS - CODE VERIFICATION

Presented in this section are sample hand calculations for comparison with printout from the computer. Two series of calculations are shown for the first time increment when:

1) the system includes no relieving capacity
2) relief capacity, equivalent to the discharge of a single PORV is included, the valve being set to open at the initial system pressure of 300 psia. NEU Millstone Unit No. 2 plant parameters are used in the example.

INITIAL CONDITIONS AND CONSTANTS

$m_m = 323425$ lbm/S6

$M_w = 191118$ lbm/S6

$A_s = 90232$ ft^2

$Q^{P1} = 224.2$ ft^3/sec

$Q^{P2} = 367$ ft^3/sec

$Q^{RC} = 187.4$ ft^3/sec

$v_0^{RP} = v_0^{RA} = 0.0161895$ ft^3/lbm

$v_0^{P1} = v_0^{P2} = 0.0167588$ ft^3/lbm

$v^{PZR} = 0.0188942$ ft^3/lbm

$v_0^{RCS} = 0.0167616$ ft^3/lbm

$m_0^{RCS} = 660139$ lbm

$h_0^{RCS} = 189.257$ Btu/lbm

$V^{P1} = V^{P2} = 1148 ft^3$

$V^{RA} = 1977 ft^3$

$V^{RC} = 892 ft^3$

$V^{RP} = 2002 ft^3$

$T_o^{S1} = T_o^{S2} = 220°F$

$T_o^{RCS} = 220.4°F = T_{AVG}^{RCS}$

$V^{RV} = 4871 ft^3$

$P_o^{RCS} = 300 psia$

$U^{SG1} = 244 \; Btu/hr\,°F\,ft^2$

$U^{SG2} = 193 \; Btu/hr\,°F\,ft^2$

9.4.1 CODE VERIFICATION, W/O VALVE DISCHARGE

The following hand calculations use a time increment of 2 seconds. Actual code time increments are 0.1 seconds. The equations as summarized in Section 9.3.9 are applied.

Therefore, at $i=1$ and for $\Delta t = 2 sec$

① $$v_1^{P1} = \frac{1148(.0167588)(.0161895)}{(224.2\times 2)(0.0167588)+(1148-224.2\times 2)(.0161895)}$$

$$v_1^{P1} = 0.01653173 \; ft^3/lbm$$

② $$T_{1_{min}}^{P1} = f(300 psia, .01653173 \; ft^3/lbm) = 186.1°F$$

③ $$\Delta Q_1^{SG1} = (244)(90232)\left[220 - 186.1\right]\frac{2}{3600} = 414646 \; Btu$$

④ $$T_{1,1}^{P1} = \frac{(414646)(.0165317)}{(1148)(1)} + 186.1 = 192.07°F$$

⑤ $$T_1^{S1} = 220 - \frac{414646}{(333,425)(.12)+(191118)(1.0)} = 218.21°F$$

⑥ $v_1^{P1} = f(300\text{ psia}, 192.07°F) = 0.0165691\ ft^3/lbm$

The increase in primary heat content is:

$$q^{IN} = m_1^{P1} c_p (T_1^{P1} - T_{1mix}^{P1}) = \frac{V^{P1}}{V_1^{P1}} c_p (T_1^{P1} - T_{1mix}^{P1})$$

$$q^{IN} = \frac{1148}{.0165691}(1)(192.07 - 186.1) = 413635\ Btu$$

This is based on the $T_1^{P1} = 192.07$ (from step 4) which resulted using the $\Delta q_1^{SG1} = f(\Delta T^{S/P})$ of step 3. However, the heat transferred due to the revised T_1^{P1} and reduced T_1^{S1} is:

$$\Delta q = U A \Delta T = (244)(90232)(218.2 - 192.1)(1)/3600$$

$$\Delta q = 319241\ Btu$$

The amount of heat transfer is ≈ 77% of that calculated in step (3). Therefore, an iteration to recalculate T_1^{P1} and T_1^{S1} must be performed. Thus, using $\Delta q_1^{SG1} = 319241$ Btu

④ $$T_1^{P1} = \frac{(319241)(.0165691)}{(1148)(1)} + 186.1 = 190.71°F$$

⑤ $$T_1^{S1} = 220 - \frac{319241}{(333425)(.12) + (191118)(1.0)} = 218.62°F$$

⑥ $v_1^{P1} = f(300\text{ psia}, 190.71°F) = 0.0165604\ ft^3/lbm$

The increase in primary heat content is:

$$q^{IN} = \frac{1148}{.0165604}(1.0)(190.7 - 186.1) = 318881\ Btu$$

The amount of heat transfered based on the new, re-calculated values of T_1^{P1} & T_1^{S1} :

$$\Delta q_1^{S61} = (244)(90232)(218.62 - 190.71)\left(\frac{2}{3600}\right)$$

$$\Delta q_1^{S61} = 341380\ Btu$$

This value is ≈ 7 % greater than the heat absorbed by the primary coolant. This, compared with the much greater difference of the previous iteration, indicates a rapid convergence. One more iteration is here presented.

Therefore, using $\Delta q_1^{S61} = 341{,}380$ Btu :

④ $$T_1^{P1} = \frac{(341380)(.0165604)}{1148(1)} + 186.1 = 191.02°F$$

⑤ $$T_1^{S1} = 220 - \frac{341380}{(333425)(.12) + (191118)(1.0)} = 218.52°F$$

⑥ $v_1^{P1} = f(300\text{ psia}, 191.02°F) = 0.0165624\ ft^3/lbm$

The increase in primary heat content is:

$$q^{in} = \frac{1148}{.0165624}(1.0)(191.02 - 186.1) = 341023 \text{ Btu}$$

The amount of heat transfer based on the new recalculated values of T_1^{P1} & T_1^{S1}:

$$\Delta q_{01}^{SG1} = (244)(90232)(215.52 - 191.02)\frac{2}{3600}$$

$$\Delta q_{01}^{SG1} = 336\,365 \text{ Btu}$$

This value is within ≈ 1.4% of the amount absorbed by primary coolant. Thus the iteration is considered converged since the energy balance is satisfied: i.e. $\Delta q_{01}^{SG1} \approx q^{IN}$.

The new updated v_1^{P1} resulting from heat transferred from the secondary side is

$$v_1^{P1} = 0.0165624 \text{ ft}^3/\text{lbm}$$

(7) $$v_1^{RA} = \frac{1977(.0165624)(.0161895)}{(224.2\times2)(.0161895) + (1977 - 224.2\times2)(.0165624)}$$

$$v_1^{RA} = 0.01627259 \text{ ft}^3/\text{lbm}$$

(8) $$v_1^{RC} = \frac{892(.0162726)(.0161895)}{(224.2-36.7)(2)(.0161895) + (892 - (224.2-36.7)2)(.0162726)}$$

$$v_1^{RC} = 0.0162243 \text{ ft}^3/\text{lbm}$$

(9) $v_1^{P2} = \dfrac{1148(.0162243)(.0167588)}{(36.7)(2)(.0167588) + (1148 - 36.7 \times 2)(.0162243)}$

$v_1^{P2} = 0.0167236 \ ft^3/lbm.$

(10) $T_{1_{mix}}^{P2} = f(300 \text{ psia}, 0.0167236 \ ft^3/lbm) = 215.07°F$

(11) $\Delta q_{01}^{SG2} = (193)(90232)(220 - 215.07)\dfrac{(2)}{3600} = 47697 \text{ Btu}$

(12) $T_1^{P2} = \dfrac{(47697)(.0167236)}{(1148)(1)} + 215.07 = 215.76°F$

(13) $T_1^{S2} = 220 - \dfrac{47697}{(333425)(12) + (191118)(1.0)} = 219.79°F$

(14) $v_1^{P2} = f(300 \text{ psia}, 215.76°F) = 0.0167284 \ ft^3/lbm$

The increase in primary heat content is:

$q^{IN} = m_1^{P2} c_p (T_1^{P2} - T_{1_{mix}}^{P2}) = \dfrac{V^{P2}}{v_1^{P2}} c_p (T_1^{P2} - T_{1_{mix}}^{P2})$

$q^{IN} = \dfrac{1148}{.0167284}(1.0)(215.76 - 215.07) = 47351 \text{ Btu}$

However, the amount of heat transferred based on the recalculated values of T_1^{P2} & T_1^{S2} is:

$\Delta q_{01}^{SG2} = (193)(90232)(219.79 - 215.76)\dfrac{2}{3600} = 38990 \text{ Btu}$

This value is within ≈ 78% of the amount calculated as absorbed by the primary coolant.

Therefore, further iterations to recalculate T_1^{P2} and T_1^{S2} must be performed. Using $\Delta q^{SG2} = 38990$ Btu

(12) $$T_1^{P2} = \frac{(38990)(.0167284)}{(1148)(1.0)} + 215.07 = 215.64°F$$

(13) $$T_1^{S2} = 220 - \frac{38990}{(333425)(.12)+(191118)(1.0)} = 219.83°F$$

(14) $$v_1^{P2} = f(200psia, 215.64°F) = 0.0167276\ ft^3/lbm$$

Now, the increase in primary heat content is:

$$q^{IR} = \frac{1148\ (1.0)}{.0167276}(215.64 - 215.07) = 39119\ Btu$$

The amount of heat transferred based on the recalculated values of T_1^{P2} and T_1^{S2} is:

$$\Delta q^{SG2} = (193)(90232)(219.83 - 215.64)\frac{2}{3600} = 40538\ Btu$$

This value is within 4% of the amount calculated as absorbed by the primary coolant.

Therefore, the iteration is considered converged ($\Delta q_{01}^{SG2} \approx q^{IN}$). The new updated v_1^{PZ} resulting from heat transferred from the secondary to the primary is:

$$v_1^{PZ} = 0.0167276 \ ft^3/lbm$$

(15) $$v_1^{RP} = \frac{2002(.0162243)(.0167276)(.0161895)}{(224.2-36.7)(2)(.0167276)(.0161895) + (36.7\times2)(.0161895)(.0162243) + (2002-224.2\times2)(.0167276)(.0162243)}$$

$$v_1^{RP} = 0.01621514 \ ft^3/lbm$$

(16) $m_1^{out} = 0$ since $P^{RCS} \neq P^{SST}$

(17) $\Delta q_{01}^{out} = 0$ since $m_1^{out} = 0$

(18) $\Delta q_{01}^{RCS} = 336365 + 40538 + (30556)(2) = 438015 \ Btu$

heat input from RCP, PZR heaters + 1% Decay heat

(19) $m_1^{RCS} = 660139 \ lbm$

(20) $$\Delta h_1^{RCS} = \frac{(438015) + (0)}{660139} = 0.663 \ Btu/lbm$$

(21) $$T_1^{RCS} = \frac{0.663}{1.0} + 220.4\,^\circ F = 221.06$$

(22) $$v_1^{RCS} = \frac{11065}{660139} = 0.0167616 \ \frac{ft^3}{lbm}$$

(23) $$P_1^{RCS} = f\left(221.06^\circ F,\ 0.0167616 \frac{ft^3}{lbm}\right) = 382.1 psia$$

Thus, the pressure computed at the end of the first iteration is: P_1^{RCS} = 382.1 psia. This value compares very closely with the value of 381.9 psia which was computed via the computer code using the same input parameters. The discrepancy is accounted for by the greater accuracy in steam properties interpolation performed by the code and greater numbers of iterations in determining steam generator heat transfer. Code output which lists the input parameters and the results for the first seconds of the transient is shown on the following page. This output should be used for verification purposes only. It is not indicative of Millstone Unit 2 transients because of the magnitude of the time step chosen ($\Delta t = 2$ sec). A 0.1 sec time increment is used in the actual computer analyses.

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MILLSTONE OVERPRESSURIZATION
RCP START WITH HOT STEAM GENERATORS
TEMPERATURE DIFFERENTIAL OF 100 F
SINGLE PORV SET AT 2400 PSIA

INPUT VARIABLE NAME	VALUE	UNITS	DIAGNOSTI
PRINT FREQUENCY	1	ITERATIONS PER PRINT	
PRESSURIZER IS IN LOOP	2		
INITIAL TIME	0.	SECONDS	
FINAL TIME	200.000	SECONDS	
TIME STEP	2.00000	SECONDS	
SG METAL CP	.120000	BTU/LBM-DEG F	
SG WATER CP	1.00000	BTU/LBM-DEG F	
PRIMARY WATER CP	1.00000	BTU/LBM-DEG F	
SG METAL MASS / SG	333425.	LBM	
SG WATER MASS / SG	191118.	LBM	
SG1 SEC COOLANT TEMP	220.000	DEG F	
SG2 SEC COOLANT TEMP	220.000	DEG F	
SG1 SEC-PRIMARY HTC	244.000	BTU/HR-FT2-DEG F	
SG2 SEC-PRIMARY HTC	193.000	BTU/HR-FT2-DEG F	
FLOW FRAC CORE-SG1	.610000		
FLOW FRAC SG-CORE-S1	.121200		
SG1 PRIM TUBE VOL	1148.00	FT3	
SG2 PRIM TUBE VOL	1148.00	FT3	
SDC MZ INSIDE VOL	4871.00	FT3	
SDC MZ OUTSIDE VOL	6134.00	FT3	
REACTOR ANNULUS VOLUME	1977.00	FT3	
REACTOR CORE VOLUME	892.000	FT3	
REACTOR UPPER PLENUM VOLUME .	2002.00	FT3	
PRESSURIZER/SURGE LINE VOLUME	1532.00	FT3	
PRESSURIZER TEMPERATURE	417.000	DEG F	
SDC MZ INSIDE TEMP	120.000	DEG F	
SDC MZ OUTSIDE TEMP	220.000	DEG F	
RCS PRESSURE	300.000	PSIA	
RCP FLOW	78102.0	CFM	
SG AREA	90232.0	FT2	
POWER VALVE UPPER PRESS TRIP	2400.00	PSIA	
POWER VALVE LOWER PRESS TRIP	2300.00	PSIA	
POWER VALVE FLOW CAPACITY ...	200.000	LBM/SEC	
SPRING VALVE UPPER PRESS TRIP	2500.00	PSIA	
SPRING VALVE ACCUMULATION ...	.100000		
SPRING VALVE FLOW CAPACITY ..	1.00000	LBM/SEC	
BACK PRESSURE	115.000	PSIA	
HEAT ADDITION RATE	.110000E+09	BTU/HR	

	TIME (SEC)	R C S PRESSURE (PSIA)	R C S ENTHALPY (BTU/LBM)	SG1 PRIM COOL TEMP (DEG F)	SG1 SEC COOL TEMP (DEG F)	SG2 PRIM COOL TEMP (DEG F)	SG2 SEC COOL TEMP (DEG F)	SG HEAT (BT
1	0.00	300.0	189.49	220.	220.	220.	220.	
2	2.00	381.9	190.18	191.	219.	216.	220.	33326
3	4.00	499.7	191.07	175.	216.	214.	220.	83327
4	6.00	633.5	192.13	167.	214.	212.	219.	140183
5	8.00	771.2	193.22	164.	211.	211.	219.	198297
6	10.00	907.0	194.29	163.	209.	210.	219.	254601
7	12.00	1037.4	195.31	163.	207.	210.	218.	307593

9.4.2 CODE VERIFICATION, WITH VALVE DISCHARGE

In order to show the effect of a relief valve discharge, the following hand calculation is presented. For ease of calculating, the valve setpoint is assumed to be at the initial system pressure of 300 psia. Normally valve setpoints are higher than the initializing pressure (typical setpoint is 465 psia).

Again, the equations summarized in Section 9.3.9 are applied. Steps 1 through 15 are identical to those presented in the previous section (9.4.1). The following steps differ since these account for valve mass and energy flux from the system. These steps are here presented, starting with step 16. The time increment used is two seconds.

16 From Section 9.3.8,

$$\dot{m}_o^{out} = 3.95\sqrt{285-100} = 53.73 \text{ lbm/sec}$$

⑰ $\Delta q_{01}^{out} = 53.73\,(393.6)(2) = 42296 \text{ Btu}$

⑱ $\Delta q_{01}^{RCS} = 336365 + 40538 + (30556)(2) - 42296$

$\Delta q_{01}^{RCS} = 395719 \text{ Btu}$

⑲ $m_1^{RCS} = 660139 - (53.73)(2) = 660032 \text{ lbm}$

⑳ $\Delta h_1^{RCS} = \dfrac{395719 + (53.73)(189.257)(2)}{660032} = 0.630 \dfrac{\text{Btu}}{\text{lbm}}$

㉑ $T_1^{RCS} = \dfrac{0.63}{1.0} + 220.4 = 221.03°F$

㉒ $v_1^{RCS} = \dfrac{11065}{660032} = 0.01676433 \dfrac{ft^3}{lbm}$

㉓ $P_1^{RCS} = f(T_1^{RCS}, v_1^{RCS}) = f\left(221.03°F, 0.01676433 \dfrac{ft^3}{lbm}\right)$

$P_1^{RCS} = 331.7 \text{ psia}$

Thus, the pressure computed at the end of the first iteration is 331.7 psia. This value compares very closely to the 330.4 psia pressure computed via the computer using identical input parameters. The discrepency is accounted for by more accurate steam properties interpolation by the computer, and in the number of iterations to determine steam generator heat transfer.

Code output which lists the input parameters and the results for the first few seconds of the transient is shown on the following page. This output should be used for verification purposes only. It is not indicative of Millstone Unit No. 2 transients since the magnitude of the time increment chosen was large (Δt = 2 seconds) and the PORV setpoint value was low (300 psia). In the actual computer analyses, a 0.1 second time increment and 465 psia valve setpoint are used.

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MILLSTONE OVERPRESSURIZATION
RCP START WITH HOT STEAM GENERATORS
TEMPERATURE DIFFERENTIAL OF 100 F
SINGLE PORV SET AT 300 PSIA

INPUT VARIABLE NAME	VALUE	UNITS	DIAGNOSTIC
PRINT FREQUENCY	1	ITERATIONS PER PRINT	
PRESSURIZER IS IN LOOP	2		
INITIAL TIME	0.	SECONDS	
FINAL TIME	200.000	SECONDS	
TIME STEP	2.00000	SECONDS	
SG METAL CP	.120000	BTU/LBM-DEG F	
SG WATER CP	1.00000	BTU/LBM-DEG F	
PRIMARY WATER CP	1.00000	BTU/LBM-DEG F	
SG METAL MASS / SG	333425.	LBM	
SG WATER MASS / SG	191118.	LBM	
SG1 SEC COOLANT TEMP	220.000	DEG F	
SG2 SEC COOLANT TEMP	220.000	DEG F	
SG1 SEC-PRIMARY HTC	244.000	BTU/HR-FT2-DEG F	
SG2 SEC-PRIMARY HTC	193.000	BTU/HR-FT2-DEG F	
FLOW FRAC CORE-SG1	.818000		
FLOW FRAC S2-CORE-S1	.121200		
SG1 PRIM TUBE VOL	1148.00	FT3	
SG2 PRIM TUBE VOL	1148.00	FT3	
SDC NOZ INSIDE VOL	8871.00	FT3	
SDC NOZ OUTSIDE VOL	6194.00	FT3	
REACTOR ANNULUS VOLUME	1977.00	FT3	
REACTOR CORE VOLUME	892.000	FT3	
REACTOR UPPER PLENUM VOLUME .	2082.00	FT3	
PRESSURIZER/SURGE LINE VOLUME	1532.00	FT3	
PRESSURIZER TEMPERATURE	417.000	DEG F	
SDC NOZ INSIDE TEMP	120.000	DEG F	
SDC NOZ OUTSIDE TEMP	220.000	DEG F	
RCS PRESSURE	300.000	PSIA	
RCP FLOW	18182.0	CFM	
SG AREA	90232.0	FT2	
POWER VALVE UPPER PRESS TRIP	300.000	PSIA	
POWER VALVE LOWER PRESS TRIP	200.000	PSIA	
POWER VALVE FLOW CAPACITY ...	53.7000	LBM/SEC	
SPRING VALVE UPPER PRESS TRIP	2500.00	PSIA	
SPRING VALVE ACCUMULATION ...	.100000		
SPRING VALVE FLOW CAPACITY ..	1.00000	LBM/SEC	
BACK PRESSURE	115.000	PSIA	
HEAT ADDITION RATE	.110000E+07	BTU/HR	

	TIME (SEC)	R C S PRESSURE (PSIA)	R C S ENTHALPY (BTU/LBM)	SG1 PRIM COOL TEMP (DEG F)	SG1 SEC COOL TEMP (DEG F)	SG2 PRIM COOL TEMP (DEG F)	SG2 SEC COOL TEMP (DEG F)	S HEAT (B
1	0.00	300.0	189.19	220.	220.	220.	220.	
2	2.00	335.8	190.10	191.	219.	216.	220.	3332
3	4.00	394.3	191.02	175.	216.	213.	220.	8379
4	6.00	467.7	192.06	167.	213.	211.	219.	14161
5	8.00	539.1	193.14	162.	211.	210.	219.	20120
6	10.00	603.2	194.21	161.	209.	209.	219.	25951
7	12.00	657.8	195.25	161.	208.	208.	218.	31503

POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

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APPENDIX A: Nomenclature

The following nomenclature applies to the calculation. Also listed are the names used in the code.

Symbol	Description	Units	Code Name
A_S	heat transfer area per SG	ft^2	AREASG
C_m^s	specific heat, secondary metal	Btu/lbm°F	CPSGM
C_w^s	specific heat, secondary water	Btu/lbm°F	CPSGW
C_w^P	specific heat, primary water	Btu/lbm°F	CPPRW
F_{RV}	RCP flow fraction through RV core to SG1	-	FRV
F_{P2}	RCP flow fraction through SG2	-	FP2
h^{RCS}	average RCS enthalpy	Btu/lbm	HRCS
Δh^{RCS}	enthalpy change in time increment	Btu/lbm	DHRCS
m^{RCS}	total mass of RCS inventory	lbm	MRCSZ
m_m^s	mass of secondary metal/SG	lbm	MSGM
m_w^s	mass of secondary water/SG	lbm	MSGW
$\dot{m}_r$	rated relief valve capacity	lbm/sec	WVALP, WVALS
$\dot{m}^{out}$	relief valve discharge	lbm/sec	MLOST
P_B	relief valve back pressure	psia	PBACK
P_{SET}	relief valve set pressure	psia	PMAXP, PMAXS
P^{RCS}	RCS pressure	psia	PRCS
Q_{P1}	SG1 throughput	ft^3/sec	WP(1)
Q_{P2}	SG2 throughput	ft^3/sec	WP(2)
Q_{RV}	reactor core throughput	ft^3/sec	WRV
Q_{RCP}	single RCP operation capacity	ft^3/min	WRCP

Appendix A - continued.....

Symbol	Description	Units	Code Name
q^{P1}	average SG1 heat transferred during time increment	Btu	QP (1)
q^{P2}	average SG2 heat transferred during time increment	Btu	QP (2)
q^{T}	integrated heat transferred	Btu	QPSUM
q^{IN}	increase in SG primary inventory heat content during time increment	Btu	HEATP
T^{RCS}	average RCS temperature	°F	TRCS
T^{H}_{SDC}	initial temperature "outside" of SDC nozzles	°F	TSDCH
T^{C}_{SDC}	initial temperature "inside" of SDC nozzles	°F	TSDCC
T^{P1}	SG1 primary temperature	°F	TP(1)
T^{S1}	SG1 secondary temperature	°F	TS(1)
T^{P2}	SG2 primary temperature	°F	TP(2)
T^{S2}	SG2 secondary temperature	°F	TS(2)
$\Delta T^{S/P}$	average sec. to prim. temperature differential	°F	-
Δt	time interval	sec	DTIME
t	integrated time elapsed during transient	sec	-
U_{SG1}	SG1 overall heat transfer coefficient	Btu/hr°Fft^2	USG(1)
U_{SG2}	SG2 overall heat transfer coefficient	Btu/hr°Fft^2	USG(2)
-	spring valve accumulation (fraction of set pressure)	-	ACCS

Appendix A - continued.....

Symbol	Description	Units	Code Name
V_{RCS}	total RCS volume	ft^3	VOLRCZ
V_{RV}	volume "inside" SDC nozzles	ft^3	VOLRV
V_{SG}	volume "outside" SDC nozzles	ft^3	VOLSG
V_{P1}	SG1 primary tube volume	ft^3	VOLP(1)
V_{P2}	SG2 primary tube volume	ft^3	VOLP(2)
ΔV_{P1}	volume displacement through SG1 during time increment	ft^3	-
ΔV_{P2}	volume displacement through SG2 during time increment	ft^3	-
υ^{P1}	ave. specific volume of SG1 primary coolant in tubes	ft^3/lbm	VP(1)
υ^{P2}	ave. specific volume of SG2 primary coolant in tubes	ft^3/lbm	VP(2)
υ_{PZR}	pressurizer specific volume	ft^3/lbm	VPZR
υ_{RCS}	average RCS specific volume	ft^3/lbm	VRCZ
υ^{RA}	reactor annulus specific volume	ft^3/lbm	VRA
υ^{RC}	reactor core specific volume	ft^3/lbm	VRC
υ^{RP}	reactor upper plenum specific volume	ft^3/lbm	VRV

Note: Symbol subscripts and superscripts were chosed from the following considerations:

1) Superscripts indicate a time variant parameter; the particular time increment is indicated for these variables by subscripts such as "i", "i-1", and "i + 1". Initial conditions are indicated by a zero subscript designation.

2) Parameters which remain constant throughout the calculation use only subscript designations.

MISC-PEC-281

COMPUTER CODE CERTIFICATE

The following code, as noted by its name, version number, and permanent file identification, is hereby approved for design application.

Code Name OVERP

Version Number 0

Permanent File Identification OVERP

Computer CDC 7600

Code Testing Completed By S.E. Weismantel SEWeismantel Date 10/9/77

Reviewed By W.B. O'Connell W.B. O'Connell Date 10/18/77

ENGINEERING AND DEVELOPMENT
QUALITY ASSURANCE OF DESIGN PROCEDURE

Calculation Number MISC-PEC-396

Figure 6-2

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EFFECTIVE April 2, 1979	
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EXHIBIT No. 5.2-4	

COMPUTER CODE CERTIFICATE

The following code, as noted by its name, version number, and permanent file identification, is hereby approved for design application.

Code Name OVERP

Version Number 01

Permanent File Identification OVERP1LGO

ID= WOLPERT

Computer CDC 7600

Code Testing Completed By M.J. Wolpert III Date 5-31-83

Reviewed By R. J. Cashkoun Date 6-1-83

COMPUTER CODE CERTIFICATE

The following code, as noted by its name, version number, and permanent file identification, is hereby approved for design application.

Code Name	OVERP1
Version Number	01/PC
Permanent File Identification	OVERP1.EXE 1-17-86 10:36 am
Computer	IBM PC

Code Testing Completed by M.J. Wolpert III (signed) 1-17-86
M.J. Wolpert III — Date

Reviewed by R.F. Paakkonen (signed) 1-21-86
R.F. Paakkonen — Date

Figure 6-1