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SCE&G®
A SCANA COMPANY

June 22, 2005
RC-05-0090

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTN: Mr. Robert E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST
REACTOR COOLANT SYSTEM – HEATUP/COOLDOWN CURVES

Reference: Jeffrey B. Archie, SCE&G, to NRC, RC-04-0170, October 22, 2004

Pursuant to 10 CFR 50.90, South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS).

The proposed changes will revise the heatup and cooldown curves located in TS section 3/4.4.9 Reactor Coolant System Pressure/Temperature Limits and the associated Technical Bases. These changes are required based on the analysis and results of the last reactor vessel surveillance specimen that was removed and analyzed as detailed in the above referenced letter. The requested changes result in curves that are more stringent than those currently in the VCSNS TS. In accordance with Administrative Letter 98-10, actions have been implemented to procedurally revise acceptance criteria and incorporate the more stringent requirements into the applicable procedures as an administrative control.

Enclosed as Attachment IV is WCAP-16305-NP, Revision 0, "V. C. Summer Heatup and Cooldown Limit Curves for Normal Operation", dated August 2004. This report details the development of the revised heatup and cooldown curves based on the data obtained from the analysis of the surveillance capsules removed from the VCSNS reactor vessel and updated calculated neutron fluences. The new VCSNS curves were generated using the "axial flaw" methodology of the 1998 ASME Code, Section XI through the 2000 Addenda. Included in this methodology is the use of the K_{Ic} stress intensity factors that were formally documented under ASME Code Case N-641.

The curves provided by this proposed change do not include instrument uncertainties. The curves that provide the operational limitations are located in plant operating procedures. Instrument uncertainties, elevation differences between the relief valves and the reactor vessel beltline, and the effect of forced flow are factored into developing the operational limitations.

Attachment I provides the TS pages marked up with the proposed changes. Attachment II provides the retyped TS pages.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed change. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91(b).

There are no other TS changes in process that will affect or be affected by this change request. There are no significant changes to any FSAR or FPER sections.

There are no commitments resulting from this change request.

If you have any questions or require additional information, please contact Mr. Ronald B. Clary at (803) 345-4757.

I certify under penalty of perjury that the foregoing is true and correct.

6/22/05
Executed on

Jeffrey B. Archie

AMM/JBA/mb

Enclosure:
I. Evaluation of the proposed change

Attachments:
I. Proposed Technical Specification Change - Mark-up
II. Proposed Technical Specification Change - Retyped
III. List of Regulatory Commitments
IV. WCAP-16305-NP, Revision 0

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File (813.20)
DMS (RC-05-0090)

LICENSE AMENDMENT REQUEST
TECHNICAL SPECIFICATION 3/4.4.9 AND ASSOCIATED BASES

1.0 DESCRIPTION

South Carolina Electric & Gas Company (SCE&G) requests an amendment to revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specification (TS) Section 3/4.4.9, Figures 3.4-2 and 3.4-3, and the associated Bases Section 3/4.4.9. The proposed change will:

a) Update Figures 3.4-2 and 3.4-3 to incorporate the latest available reactor vessel analysis and updated calculated fluences. The figures are applicable up to 56 Effective Full Power Years (EFPY).

b) Revise Bases 3/4.4.9 to delete outdated information and provide revised information applicable to the updated heatup/cooldown curves. The outdated information is being deleted and the revised information is contained in the referenced Westinghouse Topical Report. It was determined that this level of technical data is not required in the VCSNS TS Bases.

2.0 PROPOSED CHANGE

Specifically the proposed changes would revise the following:

2.1 TS Figures 3.4-2 and 3.4-3

These figures provide the VCSNS Reactor Coolant System heatup and cooldown limitations. The proposed changes incorporate the latest available reactor vessel information and updated calculated fluences, and also extend the applicability of the figures up to 56 EFPY. These proposed curves are more limiting than those currently contained in the TS.

2.2 TS Bases 3/4.4.9

The Bases are being updated to delete information that is no longer applicable and provide the reference to the Westinghouse Topical Report that contains the updated information.

3.0 BACKGROUND

Pressure/Temperature (P/T) limits are developed to satisfy 10 CFR Part 50, Appendix A, Design Criteria 14 and 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, and of rapid or gross failure. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The requirements of 10 CFR 50, Appendix G, Fracture Toughness Requirements, describe the requirements for developing P/T limits and the basis for the limitations. The proposed VCSNS P/T limit curves were generated using the "axial flaw" methodology of the 1998 edition of the ASME Code, Section XI through the 2000 Addenda. Included in this methodology is the use of the K_{IC} stress intensity factors which were formally documented under ASME Code Cases N-640 and N-641. The method to predict the reactor vessel material irradiation damage is provided in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials". Regulatory Guide 1.99, Revision 2, was used for calculation of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The proposed P/T curves were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves". This methodology determines pressure-temperature limiting curves in accordance with the requirements of Appendix G, 10 CFR Part 50, as augmented by Appendix G, Section XI of the ASME Code.

The proposed P/T curves were generated based on the latest available reactor vessel specimen capsule analysis results and updated calculated fluences. This reactor vessel information was previously provided to the NRC in WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program" by letter from Jeffrey B. Archie, SCE&G, to the NRC, dated October 22, 2004.

4.0 <u>TECHNICAL ANALYSIS</u>

The proposed changes to the P/T curves reflect the results of the analyses performed on the reactor vessel surveillance capsules, most recently specimen Z, as part of the reactor vessel material irradiation surveillance specimen program. The analysis was performed and the calculations prepared using guidance contained within Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", and Appendix G to 10 CFR 50. Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels used for light-water-cooled reactor vessels.

The results of these analyses of specimen Z were provided to the NRC in a letter from Jeffrey B. Archie, SCE&G, to the NRC, dated October 22, 2004 as required by Appendix H to 10 CFR 50. The results were detailed in an attachment to that letter, WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program". Also provided at that time was WCAP-16306-NP, Revision 0, "Evaluation of Pressurized Thermal Shock for V. C. Summer".

The new P/T curves for normal heatup and cooldown of the primary reactor coolant system have been developed using the methods as discussed in the attached WCAP-16305-NP, Revision 0, "V. C. Summer Heatup and Cooldown Limit Curves for Normal Operation". These curves provide the limits for operation of the Reactor Coolant System during heatup, cooldown, criticality, and hydrostatic testing. The curves are applicable to 56 EFPY and are being incorporated within the TS to preclude the necessity for a later TS change. These proposed curves are more limiting than those currently contained in the TS. Instrument uncertainties are not included in the TS figures; however, they are incorporated into the curves located in the plant operating procedures. The operational limit curves include the following effects:

1. Instrument uncertainties associated with the pressure and temperature measurements.
2. Pressure increases due to the elevation head differences between the pressure measurement location and the reactor vessel beltline region.
3. Pressure increases between the pressure measurement location and the reactor vessel beltline region due to Reactor Coolant System flow (i.e., form, friction, and velocity head effects resulting from Reactor Coolant Pump operation).

5.0 REGULATORY SAFETY ANALYSIS

No Significant Hazards Consideration

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10CFR50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes revise the P/T limit curves to provide figures that reflect the results of the analysis performed on reactor vessel surveillance specimen Z. This analysis was performed using NRC approved methodology as documented in WCAP 14040-NP-A, Revision 4, utilizing the 1998 ASME Code, Section XI through the 2000 Addenda, Appendix G requirements. These curves provide the limits for operation of the Reactor Coolant System during heatup, cooldown, criticality, and hydrostatic testing. These curves are provided without instrument uncertainties included, however, the uncertainties are included in the curves provided in the plant operating procedures. The limits protect the reactor vessel from brittle fracture by separating the region of acceptable operation from the region where brittle fracture is postulated to occur. Failure of the reactor vessel is not a VCSNS design basis accident, and, in general, reactor vessel failure has a low probability of occurrence and is not considered in the safety analysis.

Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes revises the P/T limits curves, Section 3/4.4.9, to incorporate the results of the analysis performed on reactor vessel specimen Z. There are no physical plant design changes or significant changes in any operating procedures. This change adjusts the heatup and cooldown curves to reflect the shift in nil-ductility reference temperature of the reactor vessel as a result of neutron embrittlement. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in margin of safety?

Response: No.

The proposed changes revise the P/T limits curves, Section 3/4.4.9, to incorporate the results of the analysis performed on reactor vessel specimen Z. The new P/T curves ensure that the 10 CFR 50 Appendix G, requirements are not exceeded during normal operation including Reactor Coolant System transients during heatup, cooldown, criticality, and hydrostatic testing. The new P/T curves were prepared, using approved industry methodology, for a projected reactor vessel neutron exposure of 56 EFPY. The proposed P/T limit curves reflect a shift of the limits in a conservative direction from the current requirements. Therefore, the change does not involve a significant reduction in a margin of safety.

Pursuant to 10 CFR 50.91, the preceding analyses provide a determination that the proposed TS change poses no significant hazard as delineated by 10 CFR 50.92.

6.0 ENVIRONMENTAL CONSIDERATION

This proposed TS change has been evaluated against criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed TS change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves change to requirements with respect to inspection or Surveillance Requirements,

(i) the proposed change involves No Significance Hazards Consideration (refer to the No Significance Hazards Consideration Determination section of this TS Change Request);

(ii) there are no significant changes in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed change does not affect the generation of any radioactive effluents nor does it affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the TS incorporating the proposed change.

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Attachment to License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
B 3/4 4-6	B 3/4 4-6
B 3/4 4-14	B 3/4 4-14

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Description of Change	Reason for Change
3/4 4-31	Figure 3.4-2	Replace heatup curve	Revised P/T limits based on analysis of reactor vessel surveillance capsule.
3/4 4-32	Figure 3.4-3	Replace cooldown curve	Revised P/T limits based on analysis of reactor vessel surveillance capsule.
B3/4 4-6	B3/4.4.9	Added updated design basis information.	Revised ASME Code reference.
B3/4 4-14	B3/4.4.9	Deleted unnecessary information contained with in Bases section and updated applicable Bases information.	Correct reference for P/T limit curves.

Replace with
new Figure 3.4-2

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1, -2
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107 °F
3/4T, 94 °F

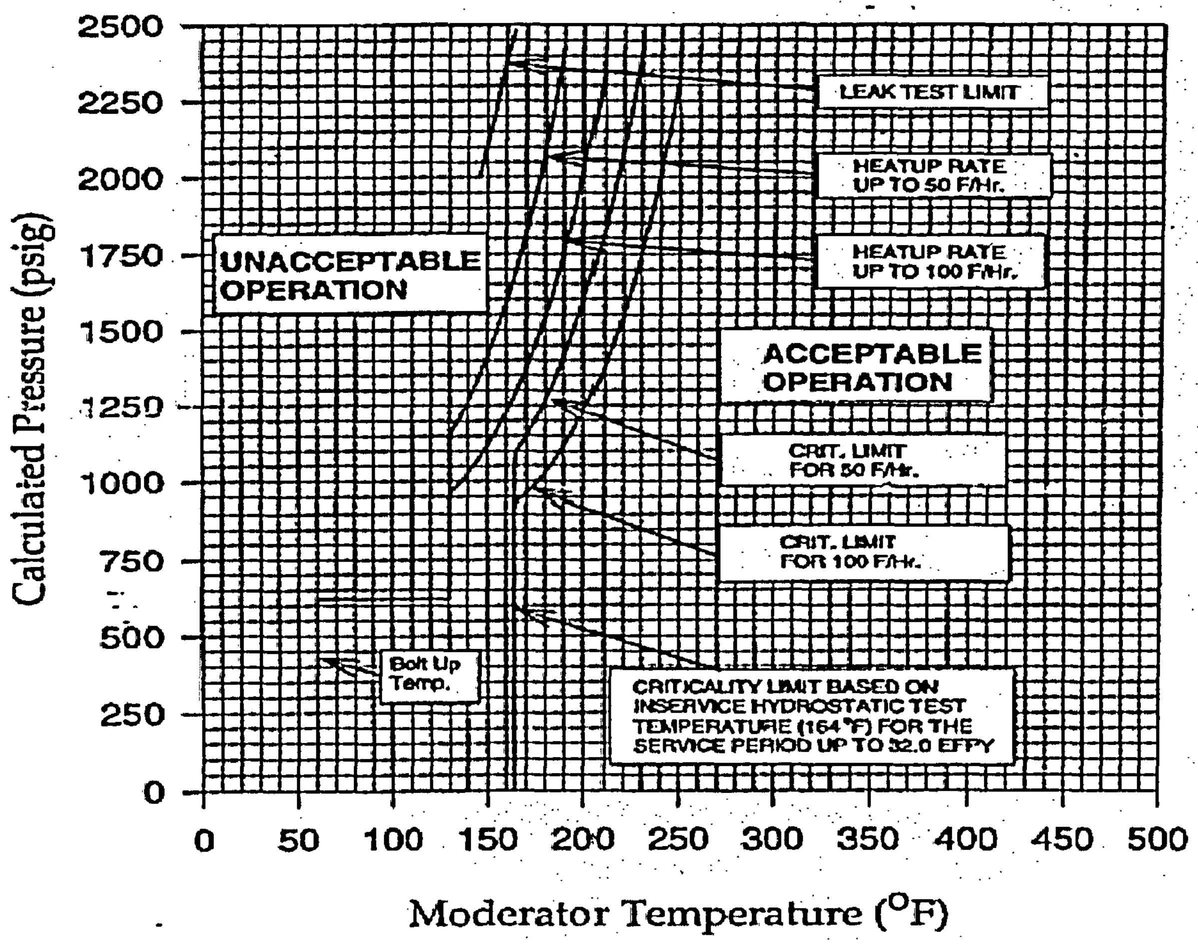

Figure 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 50 and 100 °F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS: Limiting Material: Intermediate Shell Plate A9154-1

Limiting ART Values @ 56 EFPY: 1/4T: 153°F, 3/4T: 138°F

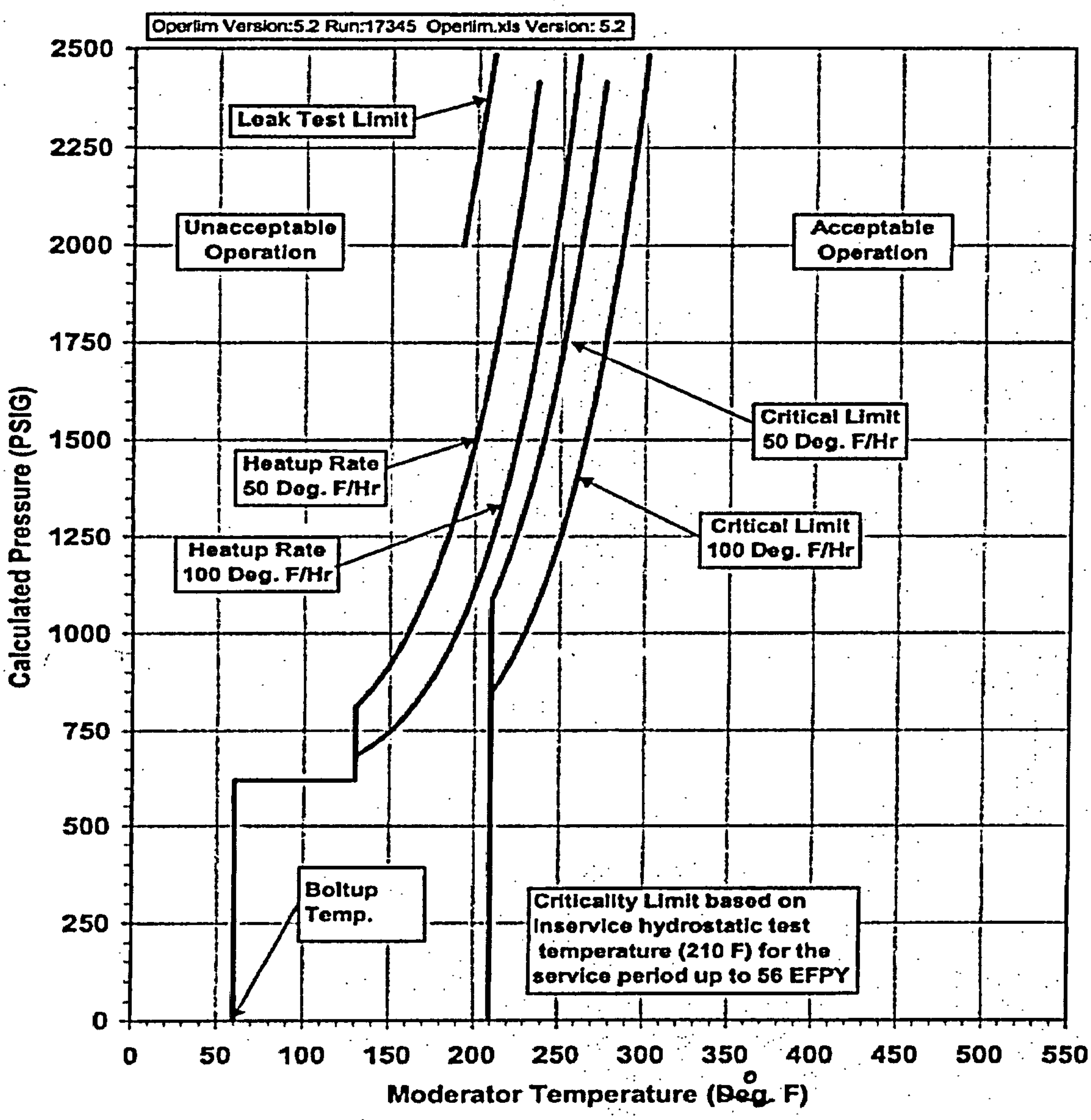

Figure 3.4-2 Unit 1 V. C. Summer Reactor Coolant System Heatup Limitations (Heatup Rates of 50 and 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

Replace with new
Figure 3.4-3

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1, -2
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107 °F
3/4T, 94 °F

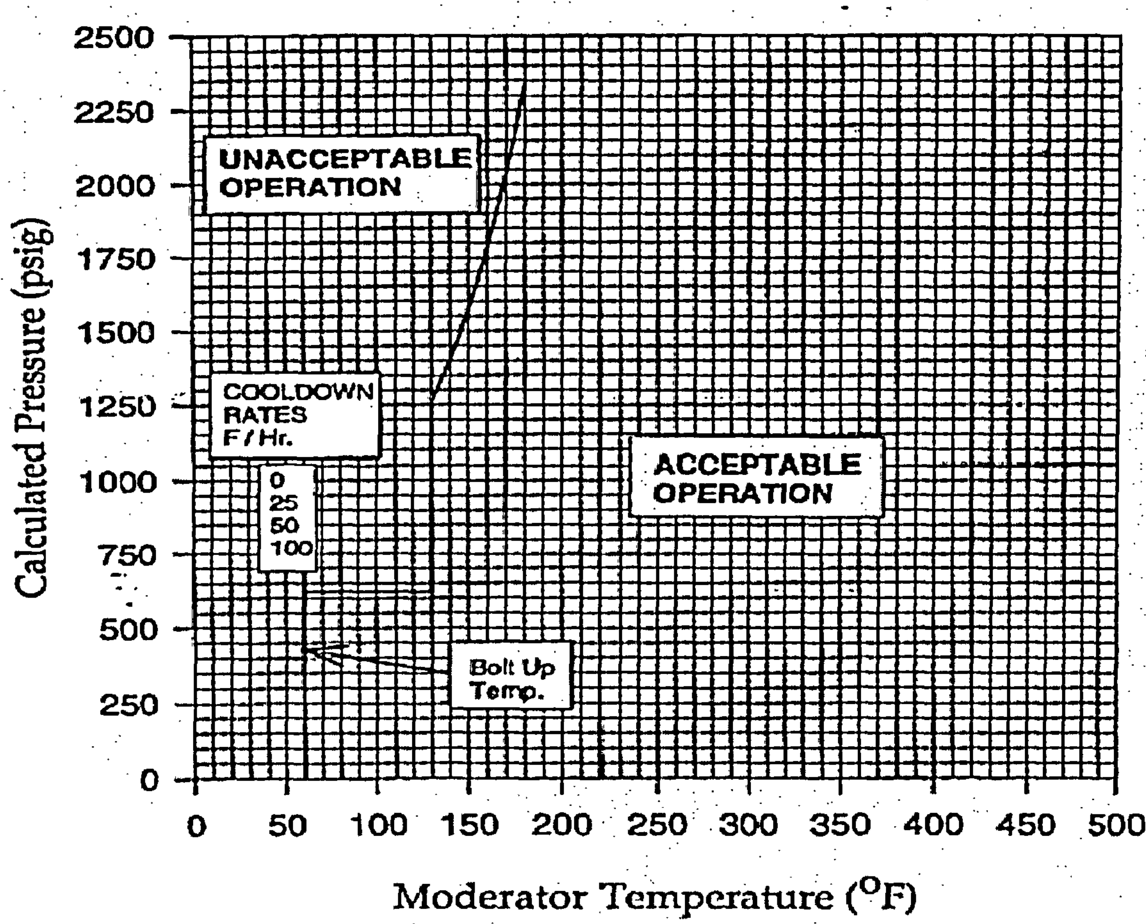

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100 °F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS: Limiting Material: Intermediate Shell Plate A9154-1

Limiting ART Values @ 56 EFPY: 1/4T: 153°F, 3/4T: 138°F

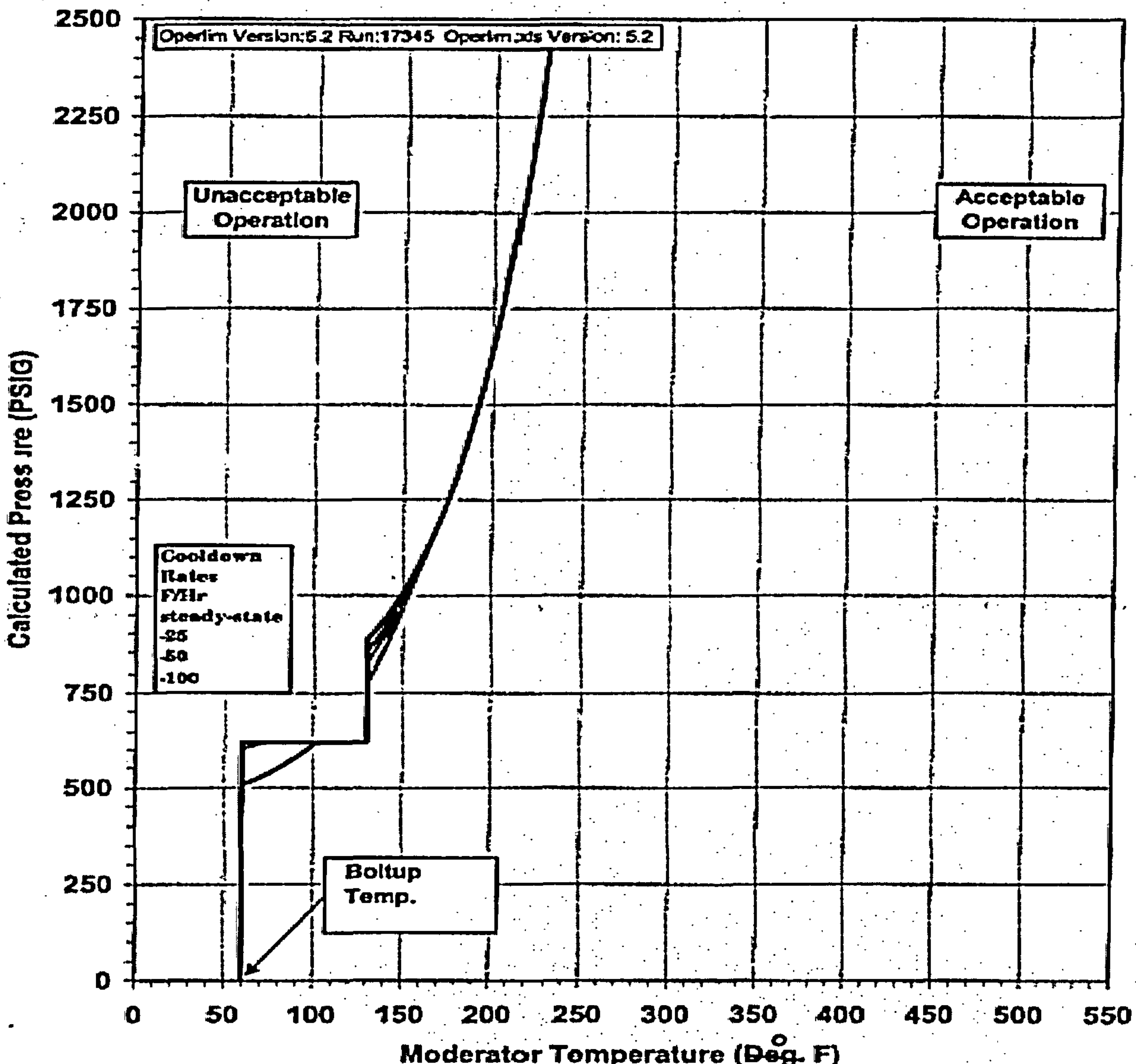

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

REACTOR COOLANT SYSTEM

BASES

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation 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 corrective 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

The temperature and pressure changes during heatup and cool down are limited by curves developed using the methodology from Westinghouse Topical Report, WCAP-14040-NP-A, updated to include the requirements of the ~~1989~~ ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, along with ASME Code Case N-~~640~~.
(through the 2000 Addenda (1998 641.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated as described in Westinghouse Topical Report, WCAP-~~15102, Rev. 2~~, "V. C. Summer ~~Unit 1~~ Heatup and Cooldown Curves for Normal Operation". 16305 - NP Revision 0

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. Charpy test specimens from Capsule W indicate that the core region lower shell plate code no. C9923-1, -2 are the limiting beltline materials for all heatup and cooldown curves to be generated. These materials exhibit limiting ART values of 107°F at 1/4T and 94°F at 3/4T at a calculated inner surface fluence of 3.84 x 10^{19} n/cm^2 at 32 EFPY.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown on the heatup and cooldown curves. The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 3.4-2. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The leak test limit curve shown in Figure 3.4-2 represents minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of the Standard Review Plan, Chapter 5.3.2 and Appendix G of the ASME Code, Section XI.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 3.4-2. The criticality limit curve specifies pressure - temperature limits for core operation to provide additional margin during actual power production as specified in Appendix G to 10 CFR 50. The pressure - temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure - temperature curve for heatup and cooldown calculated as described in this technical basis. The vertical line drawn from these points on the pressure - temperature curve, intersecting a curve 40°F higher than the pressure - temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 3.4-2 and 3.4-3 define limits for insuring prevention of nonductile failure.

The instrument uncertainties, effects of forced flow from the reactor coolant pumps, and the elevation effect of the pressure sensors are incorporated into the curves located in the plant operating procedures.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE A9154-1

LIMITING ART VALUES @ 56 EFPY: 1/4T, 153°F

3/4T, 138°F

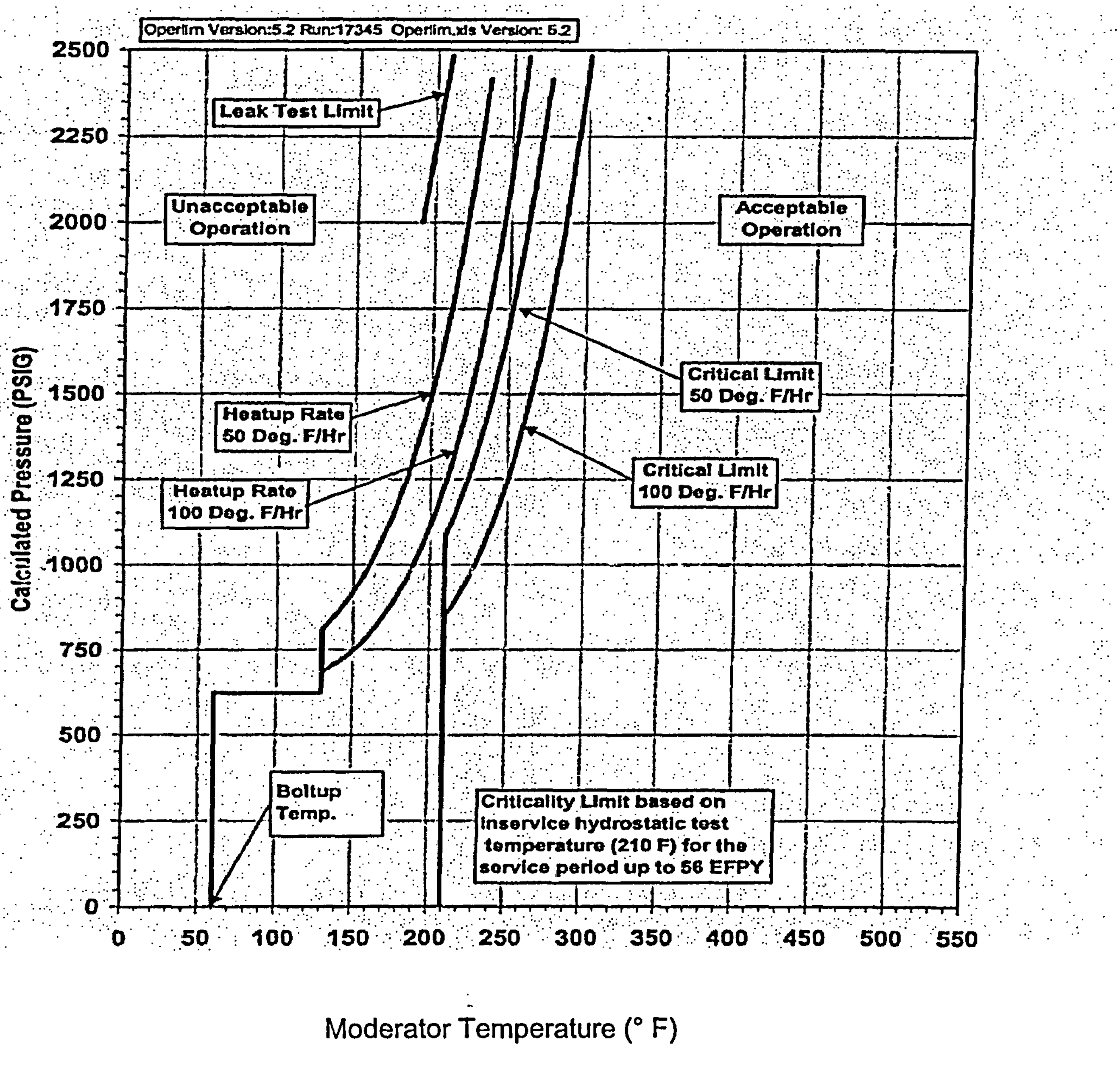

FIGURE 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 50 and 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

REACTOR COOLANT SYSTEM

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE A9154-1

LIMITING ART VALUES @ 56 EFPY: 1/4T, 153°F

3/4T, 138°F

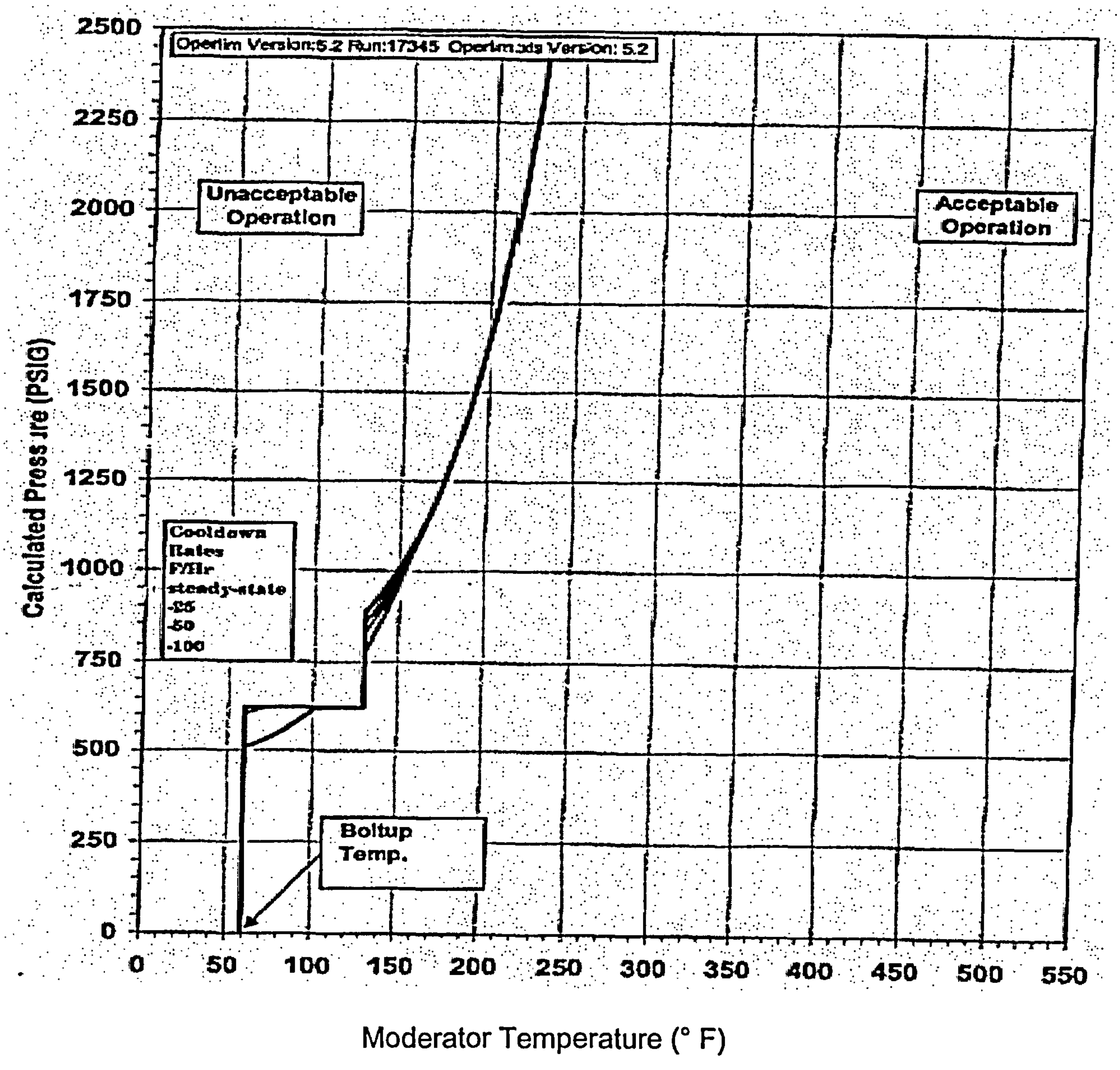

FIGURE 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 56 EFPY (Without Margins for Instrumentation Errors) Using 1998 Appendix G Methodology

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation 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 corrective 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

The temperature and pressure changes during heatup and cool down are limited by curves developed using the methodology from Westinghouse Topical Report, WCAP-14040-NP-A, updated to include the requirements of the 1998 ASME Boiler and Pressure Vessel Code, Section XI, through the 2000 Addenda, Appendix G, along with ASME Code Case N-641.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

    a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

PRESSURE/TEMPERATURE LIMITS (Continued)

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated as described in Westinghouse Topical Report, WCAP-16305-NP, Revision 0, "V. C. Summer Heatup and Cooldown Curves for Normal Operation".

ATTACHMENT III
LIST OF REGULATORY COMMITMENTS

There are no regulatory commitments created due to this License Amendment Request.

ATTACHMENT IV
WCAP-16305-NP, Revision 0

Westinghouse Non-Proprietary Class 3

WCAP-16305-NP
Revision 0

August 2004

V. C. Summer
Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse