LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

Controlling material: Base metal

Copper content: ~~Conservatively assumed to be 0.10 WT% (actual content = 0.06 WT%)~~ 0.06WT%

RT_{NDT} initial: 40°F

RT_{NDT} after ~~16~~ 11.1 EFPY: 1/4T, ~~110°F~~ 108°F

3/4T, ~~87°F~~ 86°F

Curve applicable for heatup rates up to 60°F/hr for the service period up to ~~16~~ 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

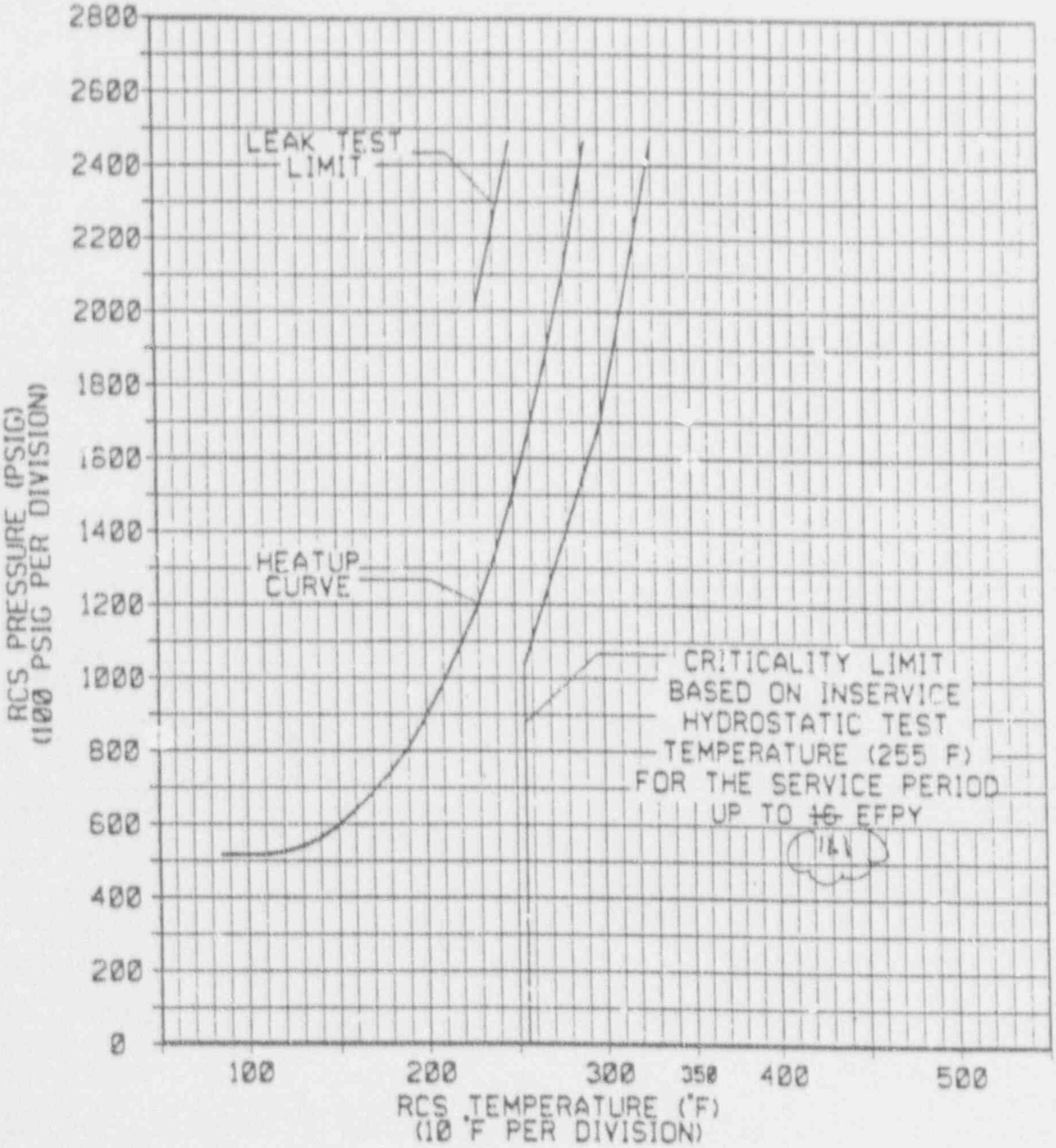

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO ~~16~~ 11.1 EFPY

MATERIAL PROPERTY BASIS

Controlling material:	Base metal
Copper content:	~~Conservatively assumed to be 0.10 WT% (actual content = 0.06 WT%)~~ 0.06 WT%
RT_{NDT} initial:	40°F
RT_{NDT} after ~~16~~ 11.1 EFPY:	1/4T, ~~110°F~~ 108°F
	3/4T, ~~87°F~~ 86°F

Curve applicable for cooldown rates up to [illegible]°F/hr for the service period up to ~~16~~ 11.1 EFPY and contains margins of 10°F and [illegible] psig for possible instrument errors

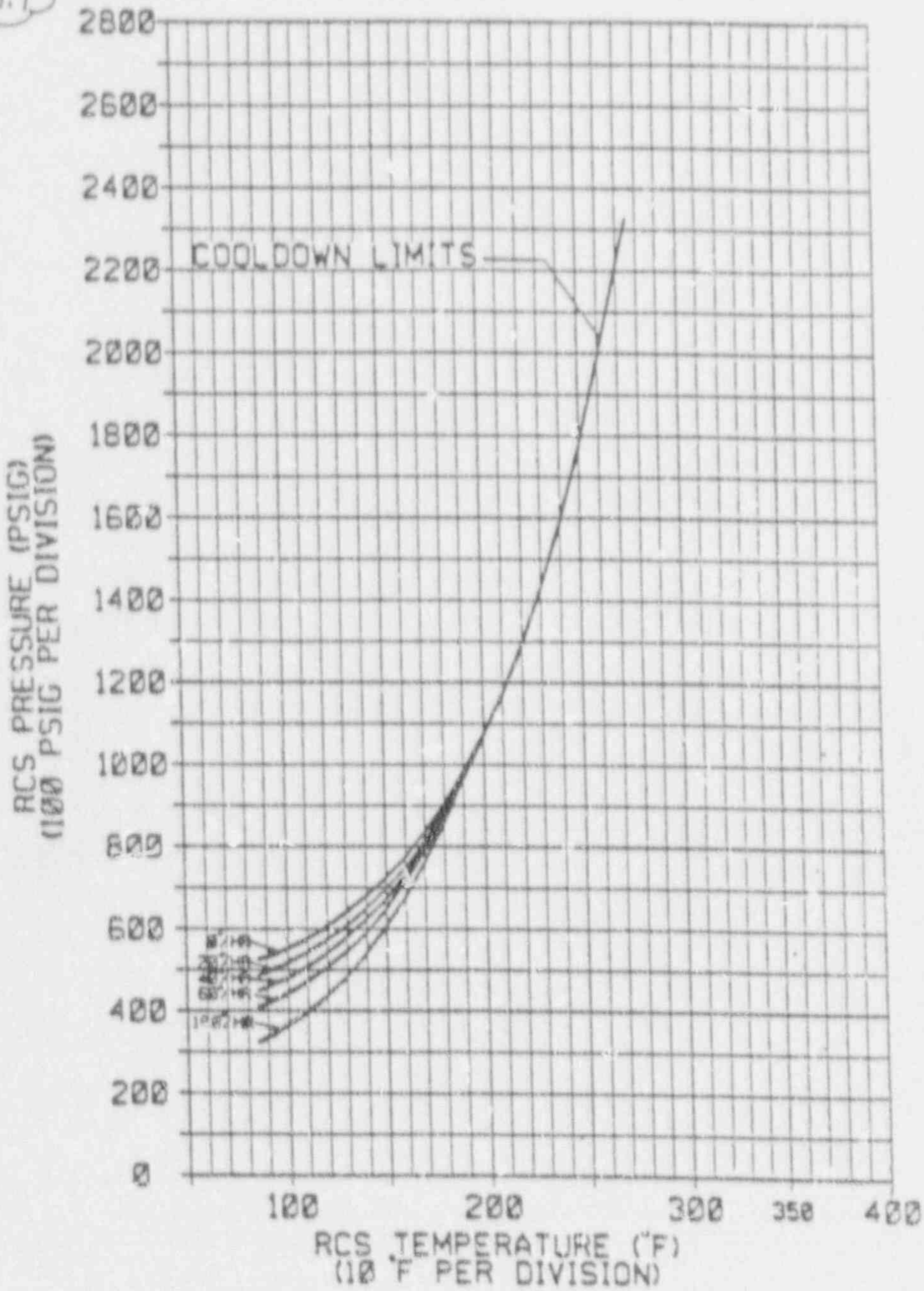

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO ~~16~~ 11.1 EFPY

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

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 for the service period specified thereon:

   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; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Regulatory Guide 1.99, Revision 2

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC ~~Standard Review Plan, ASTM E185-73,~~ and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of ~~16~~ 11.1 effective full power years (EFPY) of service life. The ~~16~~ 11.1 EFPY service life period

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

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 greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Insert 7 ~~Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.~~

~~Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-72 10 CFR Part 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.~~

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T

Insert 1

Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the mater..l in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 11.1 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. Evaluation of surveillance capsule data will be conducted in accordance with NRC Regulatory Guide 1.99, Revision 2.

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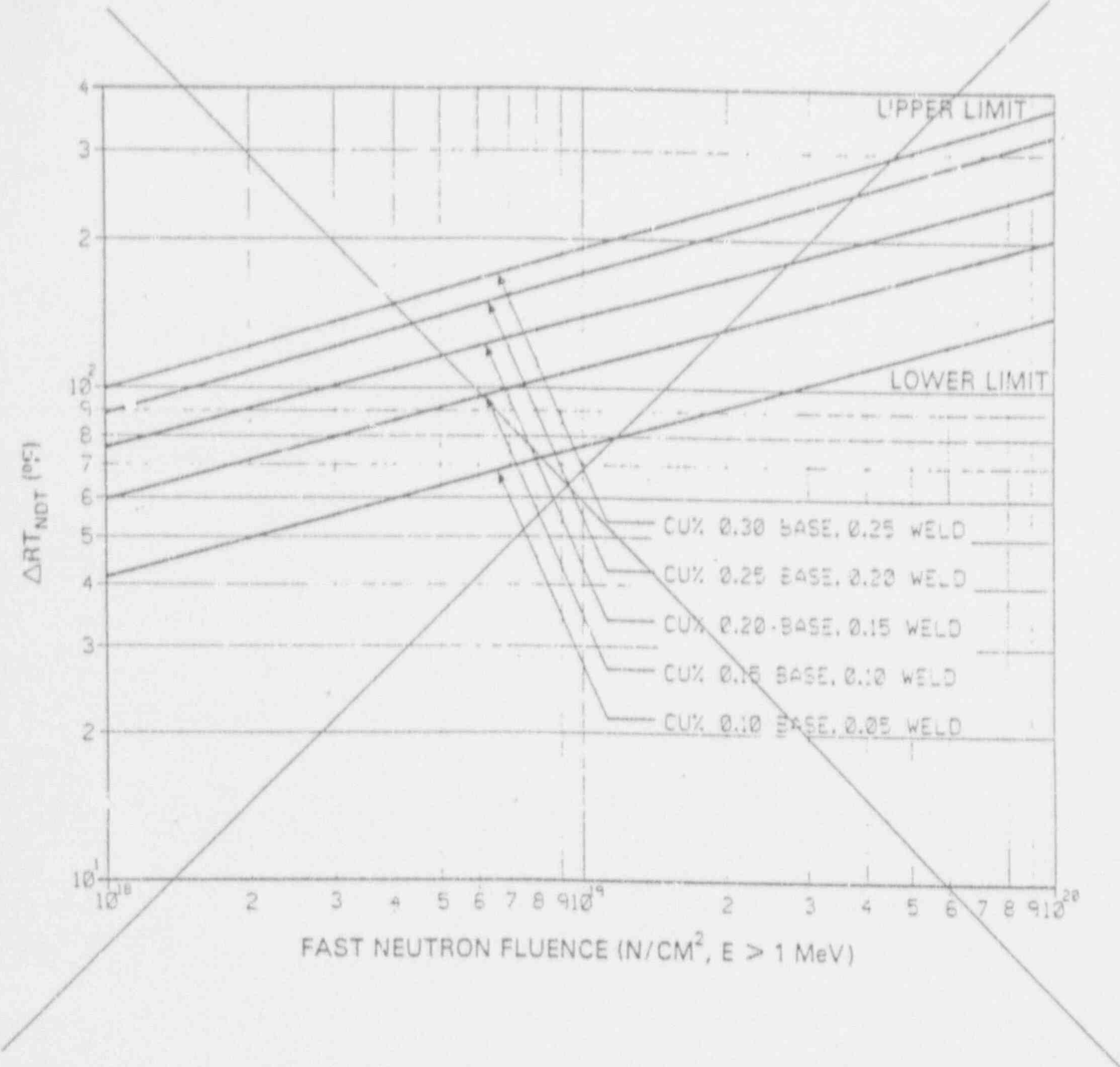

~~FIGURE B 3/4.4-2~~

~~EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F~~

III. Retype of Proposed Changes

See attached retype of proposed changes to Technical Specifications. The attached retype reflects the currently issued version of Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

Revision bars are provided in the right hand margin to designate a change in the text. No revision bars are utilized when the page is changed solely to accommodate the shifting of text due to additions or deletions.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0/4.0 BASES

Controlling material:	Base metal
Copper content:	0.06 WT%
PT_{NDT} initial:	40°F
RT_{NDT} after 11.1 EFPY:	1/4T, 108°F
	3/4T, 86°F

Curve applicable for heatup rates up to 60°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

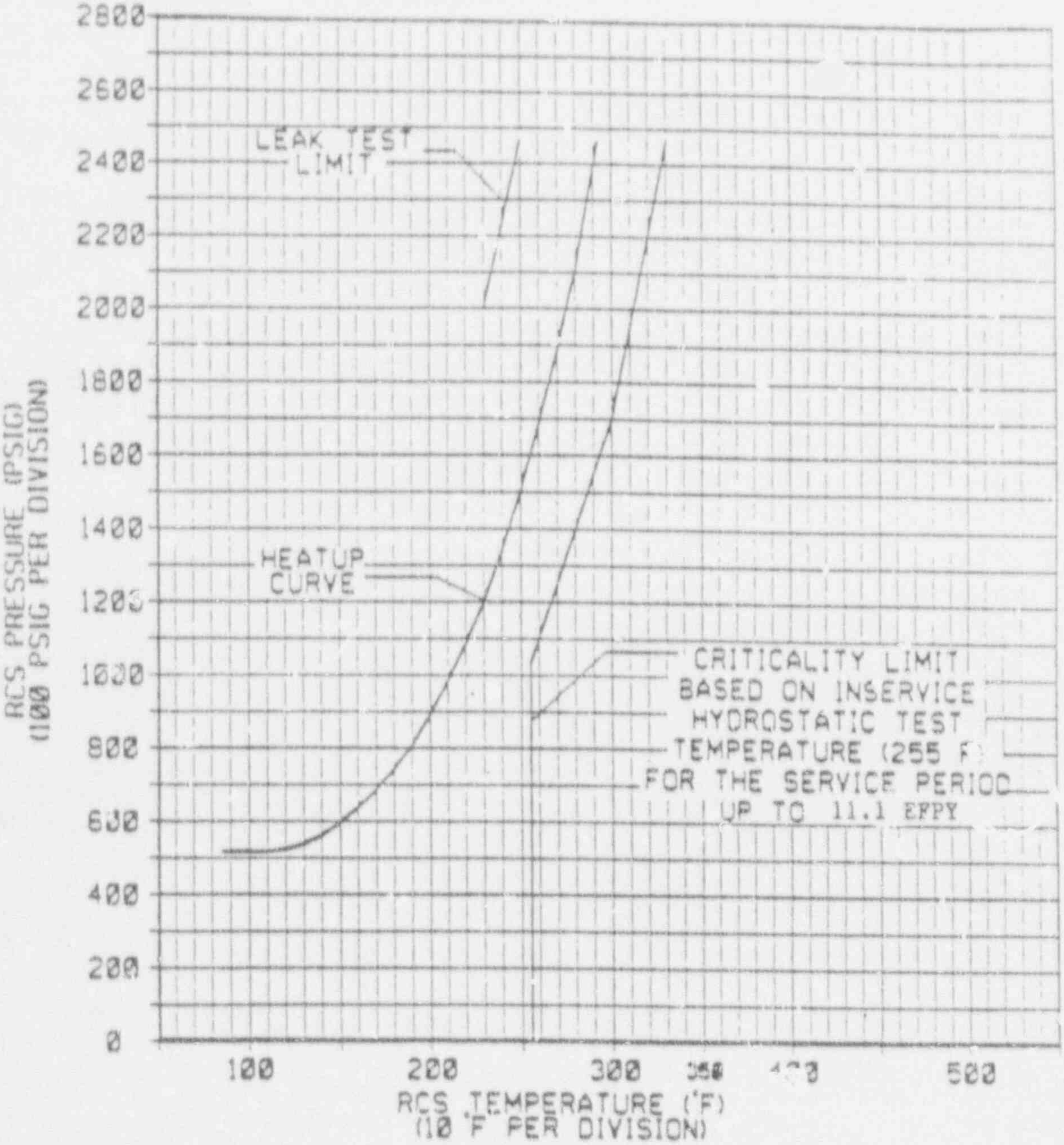

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 11.1 EFPY

Controlling material: Base metal
Copper content: 0.06 WT%

RT_{NDT} initial: 40°F
RT_{NDT} after 11.1 EFPY: 1/4T, 108°F
3/4T, 86°F

Curve applicable for cooldown rates up to 100°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

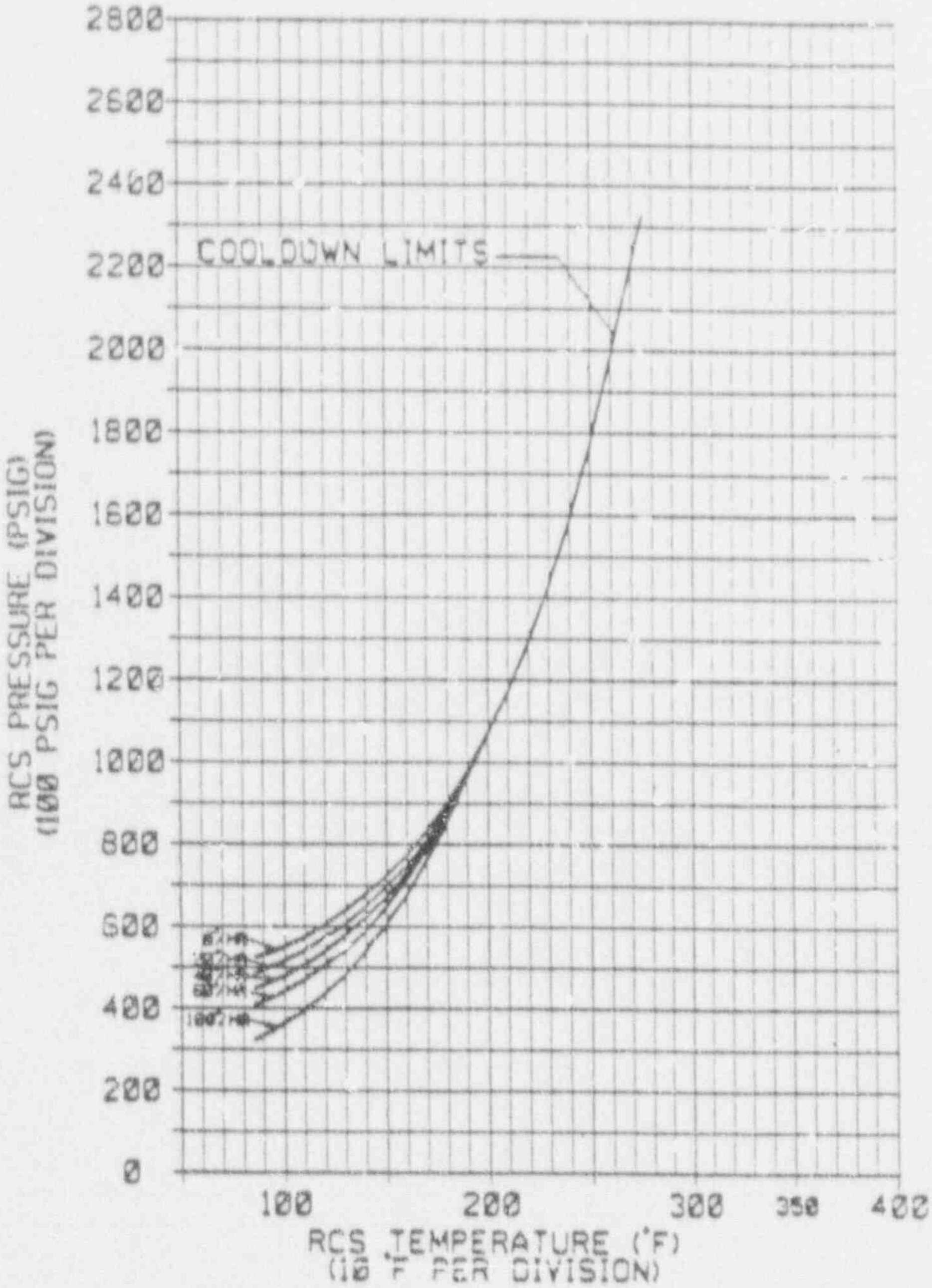

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 11.1 EFPY

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

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 for the service period specified thereon:

   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; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention ofnon-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Regulatory Guide 1.99, Revison 2, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} at the end of 11.1 effective full power years (EFPY) of service life. The 11.1 EFPY service life period

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

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 greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 11.1 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. Evaluation of surveillance capsule data will be conducted in accordance with NRC Regulatory Guide 1.99, Revision 2.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T

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IV. Safety Evaluation of Proposed Changes

Seabrook Station Technical Specification 3/4.4.9, "Pressure/Temperature Limits" are currently calculated based on the methodology described in Revision 1 to Regulatory Guide 1.99. The proposed revisions to Technical Specification 3/4.4.9, mandated by the NRC in Generic Letter 88-11, are consistent with the methodology described in Revision 2 to Regulatory Guide 1.99. Revision 2 to Regulatory Guide 1.99 describes the general procedures acceptable to the NRC staff for calculating the affects of neutron radiation embrittlement of low alloy steels currently used for light water reactor vessels. Based on calculation performed for North Atlantic by the Yankee Atomic Electric Company Nuclear Services Divisions (YNSD) utilizing the methodology described in Revision 2 to Regulatory Guide 1.99 it was concluded that current heatup and cooldown curves do not require revision if the service period is revised to 11.1 Effective Full Power Years (EFPY) versus the current 16 EFPY. The YNSD calculations are available for review at Seabrook Station. The proposed revision to Figures 3.4-2, Reactor Coolant System Heatup Limitations and 3.4-3, Reactor Coolant System Cooldown Limitations indicate that the curves are applicable 11.1 EFPY versus 16 EFPY, revises the RT_{NDT} after 11.1 EFPY at the 1/4T and 3/4T reactor vessel locations slightly downward to 108°F and 86°F respectively, and indicates that the copper content of the controlling material is 0.06WT%. This results from a more conservative approach being taken regarding radiation embrittlement of reactor vessels and does not adversely affect plant safety.

The Bases for Technical Specification 3/4.4.9 are revised to indicate that the heatup and cooldown curves are applicable for 11.1 EFPY and that the methodology described in Revision 2 to Regulatory Guide 1.99 was used to calculate the Reference Temperature for Nil Ductility Transition. Additionally, Figure B 3/4.4-2, Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550°F is deleted. The deleted figure was included in the original Seabrook Station Technical Specifications as an alternative to Regulatory Guide 1.99 for predicting the effects of radiation embrittlement on reactor vessel material. This alternative method is not used and its inclusion in Technical Specifications serves no purpose.

General Design Criterion (GDC) 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10CFR50, "Domestic Licensing of Production and Utilization Facilities", requires in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a non-brittle manner and (2) the probability of a rapidly propagating fracture is minimized. The proposed changes to Technical Specification 3/4.4.9 and its associated Bases ensure that the above criterion continues to be met.

Therefore, an adjusted reference temperature based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. Evaluation of surveillance capsule data will be conducted in accordance with NRC Regulatory Guide 1.99, Revision 2.

Generic Letter 88-11 required that revisions be made to Seabrook Station Technical Specifications based on the methodology described in Revision 2 to Regulatory Guide 1.99. The revision described herein to the Seabrook Station Technical Specifications is consistent with the methodology described in Revision 2 to Regulatory Guide 1.99. The proposed revision results in a greater degree of conservatism regarding the effects of radiation embrittlement to reactor vessels and the impact on plant operations.

V. <u>Determination of Significant Hazards for License Amendment Request 92-06 Proposed Changes</u>

(1) The proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specification 3/4.4.9 Pressure/Temperature Limits imposes a more restrictive condition on the time of applicability of the pressure/temperature operating limits curves. This revision is the result of a more conservative calculation of the affects of radiation embrittlement of reactor vessels and its impact on plant operations using the current calculational methodology delineated in NRC Regulatory Guide 1.99 Revision 2. Since the plant response to an accident will not change there is no change in the potential for a release of radiation to the public. As there is no change in the potential for an increase in the release of radiation to the public it follows that the consequences of an accident, measured in terms of dose, will not increase due to the proposed conservative revisions to the heatup and cooldown curves.

The proposed revisions to the heatup and cooldown curves do not change the function or operation of any plant equipment or effect the response of that equipment if it is called upon to operate. Since the plant will continue to function as designed there will be no significant increase in the probability of an accident previously evaluated.

(2) The proposed changes do not create the possibility of a new or different kind of accident from one previously evaluated.

The proposed revisions to the heatup and cooldown curves do not change plant design or function, effect the operation of any plant equipment or introduce any new failure mechanisms. The proposed changes to Technical Specification 3/4.4.9 provide a more conservative estimate of radiation embrittlement of the reactor vessel and reduces the time of applicability of the pressure/temperature operating limits curve. The previous accident analyses are unchanged and bound all expected plant transients and there are no new or different accident scenarios created. Therefore, the proposed revisions do not create the possibility of a new or different kind of accident from one previously evaluated.

(3) The proposed changes do not result in a significant reduction in the margin of safety.

The Bases of Technical Specification 3/4.4.9, Pressure/Temperature Limits is to assure that the Reactor Coolant System is designed and operated in a manner such that a non-ductile condition is not reached and that the Reactor Coolant System is conservatively operated in accordance with applicable Code requirements. The proposed revision is consistent with the methodology described in Revision 2 to Regulatory Guide 1.99 and does not significantly reduce the margin of safety defined in the Bases. The proposed revisions provide increased conservatism regarding radiation embrittlement of reactor vessels and its impact on plant operations.

VI. <u>Proposed Schedule for License Amendment Issuance and Effectiveness</u>

The North Atlantic response to Generic Letter 88-11 [Reference (c)] stated that the existing Seabrook Station heatup and cooldown curves are correct until eleven Effective Full Power Years and committed to submit changes to the curves utilizing the guidance of Generic Letter 88-11 and the methodology of Regulatory Guide 1.99, Revision 2 prior to the start of the second refueling outage. License Amendment Request 92-06 completes North Atlantic's commitments regarding Generic Letter 88-11.

License Amendment Request 92-06 does not propose a specific date for issuance of a license amendment. The existing Technical Specification heatup and cooldown curves are correct until 11 Effective Full Power years. If operation at a cumulative capacity factor of 80% from August 1990 is assumed, 11 Effective Full Power Years of operation will be achieved in the fourth quarter of 2003.

VII. Environmental Impact Assessment

North Atlantic has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluent that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, North Atlantic concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

VIII. Other Supporting Documentation

NHY Letter NYN-88155, "Response to USNRC Generic Letter 88-11" dated November 30, 1988, G. S. Thomas to USNRC.

PSNH

George S. Thomas
Vice President Nuclear Production

Public Service of New Hampshire

New Hampshire Yankee Division

NYN-88155

November 30, 1988

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Reference: (a) Facility Operating License No. NPF-56, Docket No. 50-443

(b) USNRC Generic Letter dated July 12, 1988, "NRC Position on Radiation Embrittl of Reactor Vessel Materials and its Impact on Plant Operations"

Subject: R ponse to USNRC Generic Letter 88-11

Gentlemen:

New Hampshire Yankee (NHY) has reviewed Reference (b) and has determined that in order to maintain compliance with Section V of 10CFR50, Appendix G, a change to the Technical Specifications will be required prior to exceeding eleven (11) Effective Full Power Years (EFPY) of operation. This change will affect Reactor Coolant System Technical Specification 3/4.4.9, and associated BASES discussion regarding Revision 2 to Regulatory Guide 1.99 computational methods for determining RT_{NDT} shift which is then used to develop plant heatup and cooldown curves. The current heatup and cooldown curves are based upon Revision 1 to Regulatory Guide 1.99 and, as stated above, are correct until eleven EFPY.

Generic Letter 88-11 requires that any required changes to Technical Specifications be submitted for approval within two refueling outages after the effective date of Revision 2 to Regulatory Guide 1.99. New Hampshire Yankee anticipates submittal of these required changes prior to its second refueling outage.

P.O. Box 300 • Seabrook, NH 03874 • Telephone (603) 474-9574

Should you require additional information please contact Mr. Robert A. Gwinn at (603) 474-9574, extension 4056.

Very truly yours,

George S. Thomas

cc: Mr. William T. Russell
Regional Administrator
Region I
United States Nuclear Regulatory Commission
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