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INDIANA MICHIGAN POWER

July 23, 2002
AEP:NRC:2349-01
10 CFR 50.90
10 CFR 50.61
10 CFR 50.60(b)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant, Unit 2
Docket No. 50-316
License Amendment Request for Unit 2 Reactor Coolant System Pressure-Temperature Curves, and Request for Exemption from Requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant, Unit 2, proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License DPR-74. I&M proposes to revise the Unit 2 reactor coolant system (RCS) pressure-temperature curves in TS Figures 3.4-2 and 3.4-3 and associated TS Bases. The revised curves will bound operation of the unit for the remainder of its current license duration and bound operation with planned license amendments to increase the power level at which the unit is allowed to operate. I&M is also proposing format changes to the affected TS pages that improve their appearance but do not affect any requirements. In support of the proposed amendment, I&M is also requesting, pursuant to 10 CFR 50.60(b), an exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G.

Enclosure 1 to this letter provides an oath and affirmation affidavit pertaining to the requested amendment. Enclosure 2 provides a description and safety analysis to support the proposed amendment, including an evaluation of significant hazards considerations pursuant to 10 CFR 50.92(c) and an environmental assessment. Attachment 1 provides TS pages that are marked to show the proposed change. Attachment 2 provides TS pages with the proposed changes incorporated. Attachment 3 provides the results of a revised fluence

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analysis of surveillance capsule U which was removed from the reactor vessel in 1992. Attachment 4 provides a description of the development of the new RCS pressure-temperature curves. Attachment 5 provides an evaluation demonstrating that the Unit 2 reactor vessel beltline materials will continue to meet the pressurized thermal shock criteria of 10 CFR 50.61. Attachment 6 contains a request for an exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G. There are no new regulatory commitments made in this letter.

Attachments 3, 4, and 5 also provide information that is applicable beyond the 32 effective full power year (EFPY) of operation assumed for the current operating license duration of 40 years. The information that is applicable beyond 32 EFPY is provided for future reference. Nuclear Regulatory Commission (NRC) review and approval of information that is applicable beyond 32 EFPY is not requested at this time.

I&M requests approval of the proposed amendment by March 1, 2003, to support implementation during the next refueling outage. Once approved, the amendment will be implemented prior to restart from that outage.

No pending amendment requests affect the TS pages that are submitted in this request. If any future submittals affect these TS pages, I&M will coordinate the changes to the pages with the NRC Project Manager to ensure proper TS page control when the associated license amendment requests are approved.

If you have any questions or require additional information, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

J. E. Pollock
Site Vice President

JW/dmb

Enclosures:
1 Affidavit
2 Evaluation of the Proposed Changes

Attachments:

1 Technical Specification Pages Marked To Show Proposed Changes
2 Proposed Technical Specification Pages
3 WCAP-13515, Revision 1, "Analysis of Capsule U from Indiana Michigan Power Company D. C. Cook Unit 2, Reactor Vessel Radiation Surveillance Program," Dated April 2002
4 WCAP-15047, Revision 2, "D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," Dated May 2002
5 WCAP-13517, Revision 1, "Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 2," Dated May 2002
6 Request for Exemption from Requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G

c: K. D. Curry, w/o Enclosures/Attachments
J. E. Dyer
MDEQ - DW & RPD, w/o Enclosures/Attachments
NRC Resident Inspector
R. Whale, w/o Enclosures/Attachments

Enclosure 1 to AEP:NRC:2612

AFFIRMATION

I, Joseph E. Pollock, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

American Electric Power Service Corporation

J E Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 23RD DAY OF July, 2002

Margaret M. Sierazl
Notary Public

My Commission Expires 11-23-2005

License Amendment Request for
Unit 2 Reactor Coolant System Pressure-Temperature Curves

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant, Unit 2, proposes to amend Appendix A, Technical Specifications (TS), of Facility Operating License DPR-74. I&M proposes to revise the Unit 2 reactor coolant system (RCS) pressure-temperature curves in TS Figures 3.4-2 and 3.4-3 and associated TS Bases. The revised curves will bound operation of the unit for the remainder of its current license duration and bound operation with planned license amendments to increase the power level at which the unit is allowed to operate. I&M is also proposing format changes to the affected TS pages that improve their appearance but do not affect any requirements. In support of the proposed amendment, I&M is also requesting, pursuant to 10 CFR 50.60(b), an exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G.

2.0 PROPOSED CHANGE

The curves in TS Figures 3.4-2 and 3.4-3 specify limits on RCS pressure and temperature for heatup, cooldown, criticality, and inservice leak and hydrostatic testing. The proposed amendment will revise these curves such that they bound operation of the reactor for up to 32 effective full power years (EFPY) at a power level of up to 3800 MWt for the current fuel cycle, Cycle 13, and beyond. The revised curves also reflect new fluence analysis methodology in accordance with Regulatory Guide (RG) 1.190 (Reference 1), reflect the use of American Society of Mechanical Engineers (ASME) Code Case N-641, include boltup limits, and do not include instrument uncertainty margins. The proposed amendment will also change the titles, labels, and orientation of TS Figures 3.4-2 and 3.4-3 and the TS Bases for Section 3/4.4.9 to be consistent with the revised curves.

I&M also proposes three types of format changes to the revised TS pages. The changes are:

Reformatting of the headers to include numbered first and second-tier TS section titles.
Reformatting of the footers to include "Page (page number)" center page, and a full-width single line to separate the footer from the page text.
Fully justifying the text and changing the font.
Changing the format of the term "Figure 3.4.2," as currently used in the TS Bases, to match the format used in TS Section 3/4.4.9, i.e., "Figure 3.4-2."

Attachment 1 to this letter provides TS pages that are marked to show the proposed changes (except for the above described format changes). Attachment 2 provides TS pages with the proposed changes incorporated.

3.0 BACKGROUND

Basis for RCS Pressure-Temperature Curves

All components in the reactor coolant pressure boundary (RCPB) 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 4.1.4 of the Updated Final Safety Analysis Report (UFSAR). The curves in TS Figures 3.4-2 and 3.4-3 establish operating limits that provide a margin to brittle failure for the RCPB, considering the effect of these cyclic loads. These curves limit the rates of temperature and pressure changes during startup and shutdown so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. These limits apply mainly to the reactor pressure vessel (RPV) since it is the component most subject to brittle failure.

The heatup limit curve provided in TS Figure 3.4-2 is a composite curve which is prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60 degrees Fahrenheit per hour. The cooldown limit curves provided in TS Figure 3.4-3 are composite curves which are prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses, while producing compressive stresses at the outside wall. The heatup and cooldown curves are based on the most limiting value of the predicted adjusted reference temperature (ART). The neutron embrittlement effect on the ART is addressed by periodically removing and evaluating one of the reactor vessel surveillance capsules and adjusting the heatup and cooldown limit curves as necessary. The current heatup and cooldown curves are based on an analysis of reactor vessel surveillance capsule U, which was removed from the vessel in 1992. These curves were approved for incorporation into the TS in 1994 (Reference 2), along with corresponding criticality and leak test curves.

Reason for Requesting Amendment

The current RCS pressure-temperature curves bound operation of the reactor for 15 EFPY at the currently licensed power level of 3411 MWt. I&M estimates that Unit 2 will reach 15 EFPY in June 2003. Additionally, I&M plans to request NRC approval of increases in the licensed power level for Unit 2. Therefore, I&M is proposing to replace the current curves with revised curves that are valid for 32 EFPY, which is expected to bound the current duration of the operating license, at a power level of up to 3800 MWt for the current fuel cycle and beyond, which will bound the planned increases in the licensed power level.

4.0 TECHNICAL ANALYSIS

The proposed amendment is supported by three analyses/evaluations that are documented in Attachments 3, 4, and 5, to this letter. Attachment 6 consists of a request for exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G, in support of Attachment 4 to this letter.

Attachment 3

Attachment 3 provides a copy of WCAP-13515, Revision 1, "Analysis of Capsule U from Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program." This WCAP documents the Charpy V-Notch testing, tensile testing, and analysis that was performed on surveillance capsule U following its removal from the RPV. Capsule U was removed in 1992 after a total of 8.65 EFPY of operation. The WCAP also documents an analysis to determine the neutron radiation environment within the RPV, including projections of future neutron exposure.

This WCAP is a revision to the original WCAP-13515 which was submitted in support of the previous amendment (Reference 2) that updated the Unit 2 RCS pressure-temperature curves. The revision includes an update of the fluence analysis methodology described in Section 6.0, "Radiation Analysis and Neutron Dosimetry." This section was revised to reflect the fluence analysis methodology specified in RG 1.190, which was issued subsequent to the original WCAP-13515. A new sub-section 6.4, "Projections of Reactor Vessel Exposure" has been added to Section 6.0. This sub-section includes fluence projections for 32 EFPY of operation, at a power level of up to 3800 MWt for the current fuel cycle and beyond. The revision to the WCAP also includes relocation of the RCS pressure-temperature curves and Table 5-7, "Projected End of License (32 EFPY) RT_{NDT} and Upper Shelf Energy Values for D. C. Cook Unit 2 Beltline Region Materials per Regulatory Guide 1.99, Revision 2." The pressure-temperature curves and table were incorporated into WCAP- 15047, Revision 2, which is provided as Attachment 4 to this letter. Finally, tables in Section 5.0, "Testing of Specimens from Capsule U," were revised to account for longitudinal and transverse directions individually as opposed to a combined table.

As documented in Section 1, "Summary of Results" of Attachment 3, the surveillance capsule materials exhibit an upper shelf energy level that is more than adequate for continued safe unit operation, and are calculated to maintain an upper shelf energy of greater than 50 ft-lb throughout the life (32 EFPY) of the vessel as required by 10 CFR 50, Appendix G.

Attachment 4

Attachment 4 provides a copy of WCAP-15047, Revision 2, "D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation." This WCAP provides the RCS pressure-temperature curves that are proposed as revisions to TS Figures 3.4-2 and 3.4-3, and describes how the curves were developed. These curves are based, in part, on fluence data from the revised analysis of surveillance capsule U documented in Attachment 3 to this letter. The fluence data includes projections for 32 EFPY, assuming operation at a power level of up to 3800 MWt for the current fuel cycle and beyond. These fluence projections have been used to calculate ART values in accordance with the guidance provided in RG 1.99 (Reference 3), as described in Section 8 of Attachment 4. The ART values were used to develop RCS pressure-temperature curves using the methodology described in Section 3 of Attachment 4.

Use of the new fluence analysis methodology specified in RG 1.190 and Code Case N-641 of Section XI of the ASME Code has resulted in pressure-temperature curves that are less restrictive than the curves provided in the current TS. Unlike the current RCS pressure temperature curves, the curves proposed as revisions to TS Figures 3.4-2 and 3.4-3 do not include a margin for instrument uncertainty. This is consistent with the NRC approved methodology documented in WCAP 14040-NP-A (Reference 4). Instrument uncertainty margins are incorporated in RCS pressure–temperature limits specified in plant procedures. Also unlike the current RCS pressure temperature curves, the curves proposed as revisions to TS Figures 3.4-2 and 3.4-3 include limits for bolting the RPV head. Inclusion of the boltup limits is consistent with WCAP 14040-NP-A.

Section 10.0 of Attachment 4, "Enable Temperature Calculation," provides the revised enable temperature limit. I&M has determined that, since the revised RCS pressure-temperature curves are less restrictive than the curves provided in the current TS, the basis for the currently licensed low-temperature overpressure protection system enable temperature remains valid.

As noted in Sections 3 and 10 of Attachment 4, the methodology of Code Case N-641 has been used in development of the RCS pressure-temperature curves and the calculation of the low temperature over-pressure protection enable temperature. I&M is requesting an exemption from requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G, to allow use of Code Case N-641. Attachment 6 to this letter documents the requested exemption and its justification.

Attachment 5

Attachment 5 provides a copy of WCAP-13517, Revision 1, "Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 2." WCAP-13517, Revision 1 is a revision to the original WCAP-13517, which was based, in part, on the fluence values in the original WCAP-13515 (Attachment 3 to this letter). The original WCAP-13517 was submitted to the NRC by Reference 5. Similarly, the pressurized thermal shock evaluation documented in Attachment 5 is based, in part, on the fluence projections documented in the current revision of Attachment 3 to this letter. As described in Attachment 5, these fluence projections have been used to determine that the pressurized thermal shock criteria of 10 CFR 50.61 will be satisfied for 32 EFPY of operation, at a power level of up to 3800 MWt for the current fuel cycle and beyond.

Summary

The technical analysis of the proposed TS changes has determined the following:

The revised RCS pressure-temperature curves will bound operation for 32 EFPY, which encompasses the current license duration, and will bound operation at a power level of up to 3800 MWt for the current fuel cycle and beyond, which will encompass planned increases in the licensed power level.

The surveillance capsule materials exhibit an upper shelf energy level that is more than adequate, and will maintain an upper shelf energy of greater than 50 ft-lb as required by 10 CFR 50, Appendix G for the currently licensed life of the unit, including planned increases in licensed power level.

The currently licensed low-temperature overpressure protection system enable temperature will remain valid since the revised RCS pressure-temperature curves are less restrictive than the curves provided in the current TS.

The pressurized thermal shock criteria of 10 CFR 50.61 are satisfied for the currently licensed life of the unit, including planned increases in licensed power level.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

I&M has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No

Probability of Occurrence of an Accident Previously Evaluated

The proposed change will revise the RCS pressure-temperature curves to bound operation of the reactor for up to 32 EFPY at a power level of up to 3800 MWt for the current fuel cycle and beyond, to reflect new fluence analysis methodology, to reflect the use of ASME Code Case N-641, to include boltup limits, and to no longer include instrument uncertainty margins.

The proposed change will not result in physical changes to structures, systems, or components (SSCs), or to event initiators or precursors. The proposed change will not affect the ability of personnel to control RCS pressure at low temperatures and, thereby, ensure the integrity of the RCPB. Use of ASME Code Case N-641 will be approved by the NRC through approval of a Donald C. Cook Nuclear Plant-specific exemption to requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G. Therefore, the proposed revision to the RCS pressure-temperature curve changes will have been determined in accordance with NRC accepted methodologies. These methodologies provide adequate assurance that the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes, and provide an acceptable level of protection against brittle failure. Additionally, the proposed changes will not impact the design or operation of plant systems such that previously analyzed SSCs will be more likely to fail. The initiating conditions and assumptions for accidents described in the UFSAR will remain as previously analyzed. Therefore, the proposed changes will not involve a significant increase in the probability of an accident previously evaluated.

Consequences of an Accident Previously Evaluated

The proposed change does not reduce the ability of any SSC to limit the radiological consequences of accidents described in the UFSAR. The proposed change will not alter any assumptions made in the analysis of radiological consequences of previously evaluated accidents, nor does it affect the ability to mitigate these consequences. No new or different radiological source terms will be generated as a result of the proposed change. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The format changes will improve the appearance of the affected pages but will not affect any requirements. In summary, the probability of occurrence and the consequences of an accident previously evaluated will not be significantly increased.

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 change will not result in physical changes to SSCs. The proposed change will not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor will it alter the design of any plant systems. The proposed change solely involves RCS pressure-temperature limits. The types of potential accidents associated with these limits have been previously identified and evaluated. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or single failures will be introduced as a result of the proposed changes. No new or different modes of failure will be created. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed RCS pressure-temperature curves will continue to provide adequate margins of protection for the RCPB. The proposed changes have been determined, through supporting analyses, to be in accordance with the methodologies and criteria set forth in the applicable regulations, or in accordance with technically adequate alternatives. Compliance with these methodologies provides adequate margins of safety and ensures that the RCPB will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events as described in the UFSAR. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will not significantly reduce the margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed changes involve no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The proposed amendment would revise TS Figures 3.4-2 and 3.4-3, and their associated Bases. The curves in these figures specify limits on RCS pressure and temperature during heatup, cooldown, criticality and inservice leak and hydrostatic testing.

The revised figures were developed based on an updated fluence analysis. The proposed amendment is also supported by an evaluation demonstrating compliance with the pressurized thermal shock screening criteria. Except as noted, the capsule analysis, curve development, and pressurized thermal shock evaluation have been conducted in accordance with 10 CFR 50.60(a), 10 CFR 50.61, 10 CFR 50, Appendix G, RG 1.190, and RG 1.99. Approval of the noted exception to 10 CFR 50.60(a) and 10 CFR 50, Appendix G, has been requested and justified in accordance with 10 CFR 50.60(b) and 10 CFR 50.12. The technical analysis supporting the proposed amendment demonstrates that it does not involve significant hazards considerations as described in 10 CFR 50.92.

Compliance with other regulations or TS will not be affected by the proposed amendment.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with NRC regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Considerations

I&M has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared concerning the proposed amendment.

7.0 Precedent Licensing Actions

The NRC has approved, by Reference 2, a previous revision to the Unit 2 RCS pressure-temperature curves based on an earlier revision of the analysis provided in Attachment 3 to this letter.

8.0 References

1. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2002

2. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit No 2 - Issuance of Amendment Re: Updated Heatup and Cooldown Curves (TAC No. M88889)," dated November 25, 1994

3. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, dated May 1988

4. WCAP 14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," dated January 1996

5. Letter from E. E. Fitzpatrick, I&M, to NRC Document Control Desk, "Updated Reference Temperature and Thermal Shock Analysis," AEP:NRC:0561F, dated April 12, 1993

6. Draft Regulatory Guide DG-1091 (Proposed Revision 13 of Regulatory Guide 1.147), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," dated December 2001

Attachment 1 to AEP:NRC:2349-01

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 2

3/4 4-25
3/4 4-26
B3/4 4-6
B3/4 4-10

See Insert 1

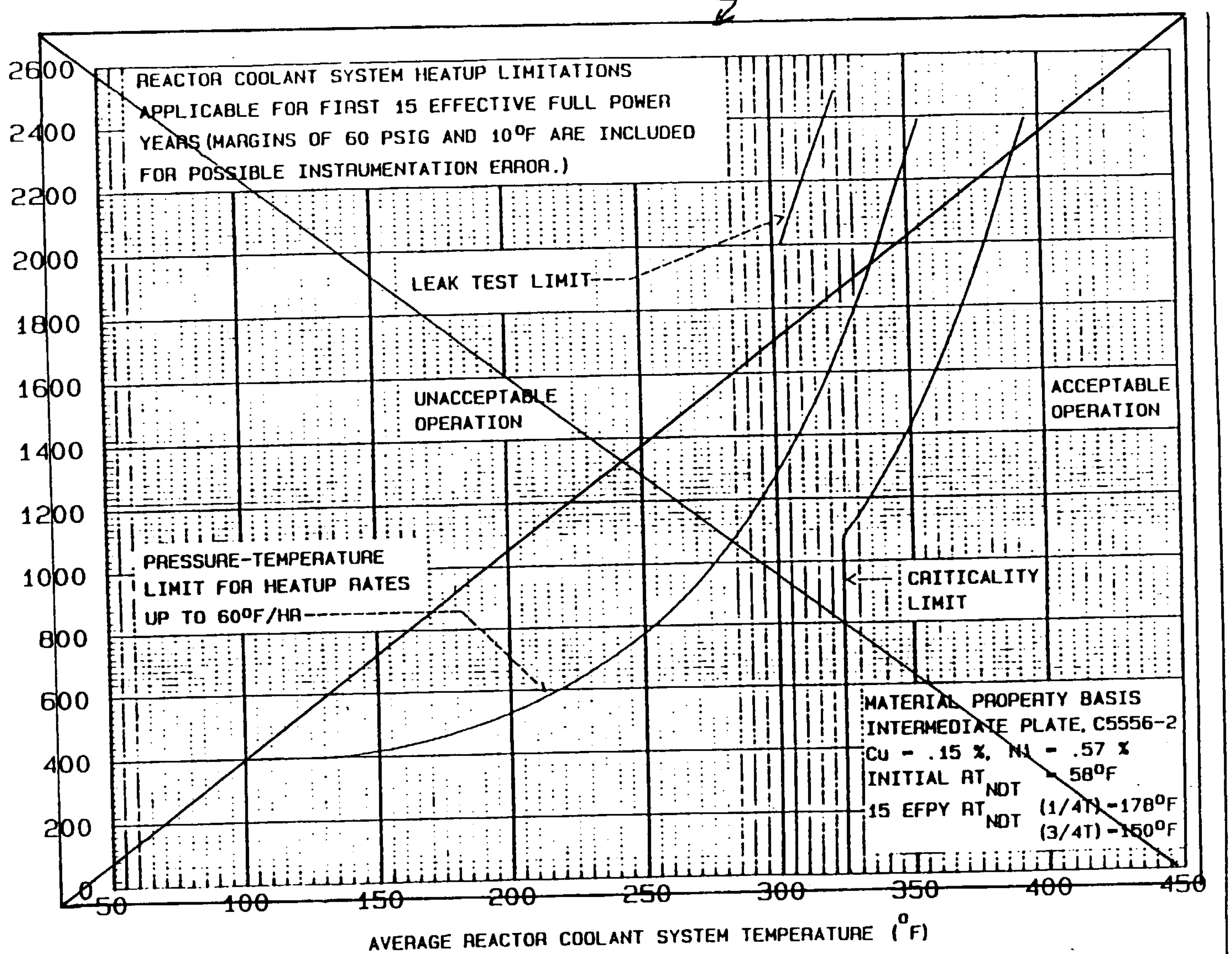
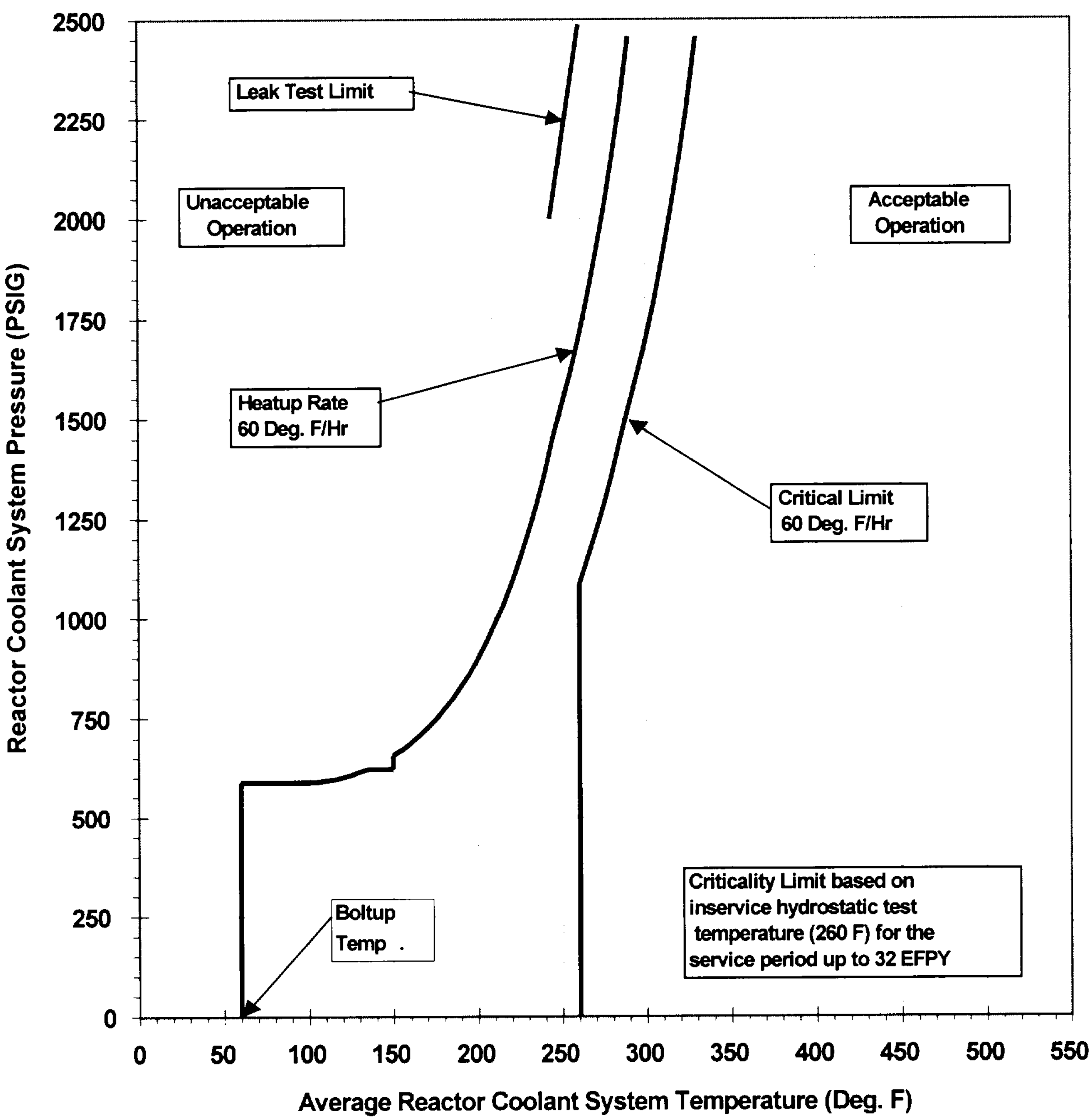


FIGURE 3.4-2

REACTOR COOLANT SYSTEM, PRESSURE - TEMPERATURE, LIMITS FOR 60 °F/HR RATE, CRITICALITY LIMIT ~~AND HYDROSTATIC TEST LIMIT~~ , BOLTUP LIMIT, AND LEAK TEST LIMIT

Insert 1

Reactor Coolant System Heatup Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F



See Insert 2

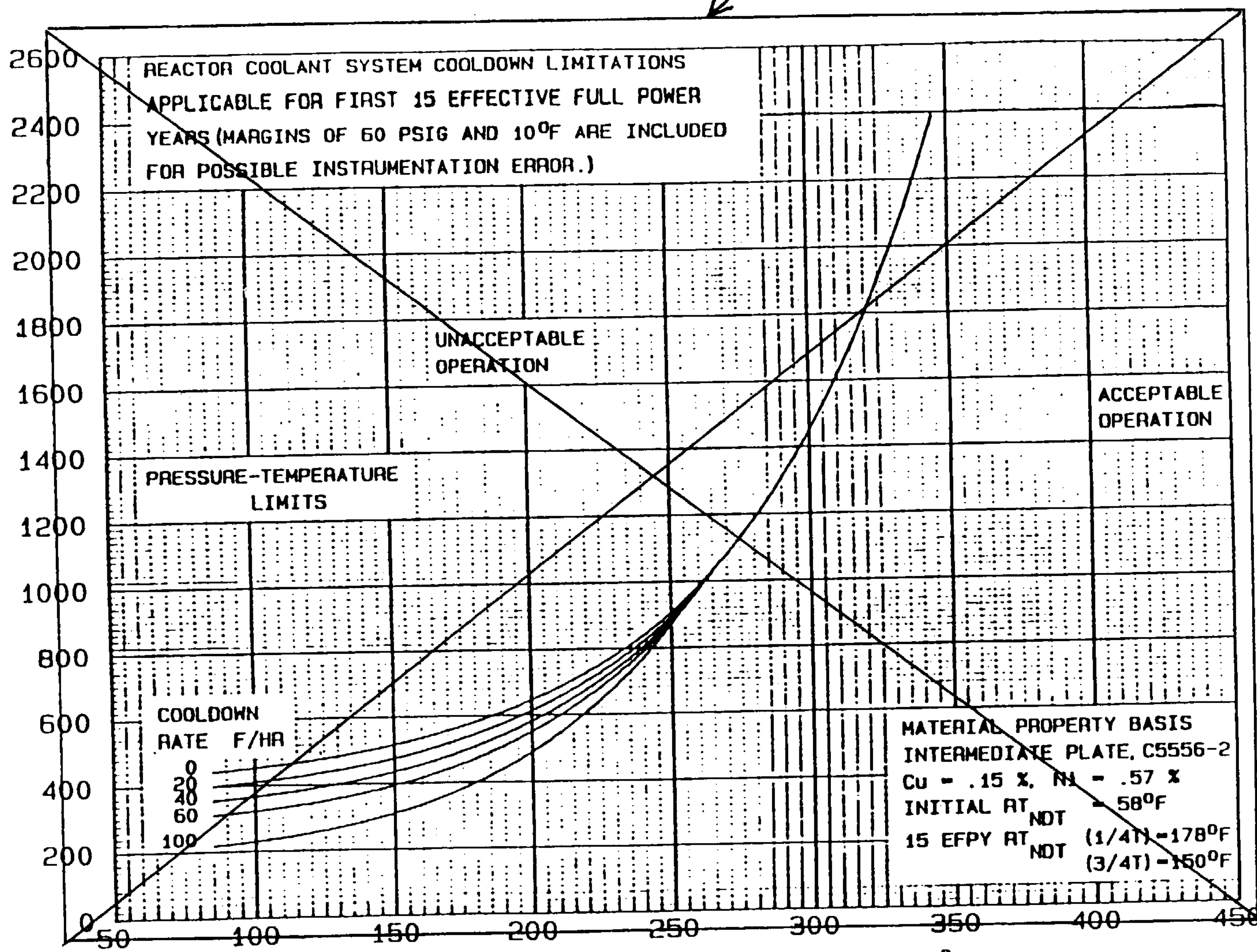
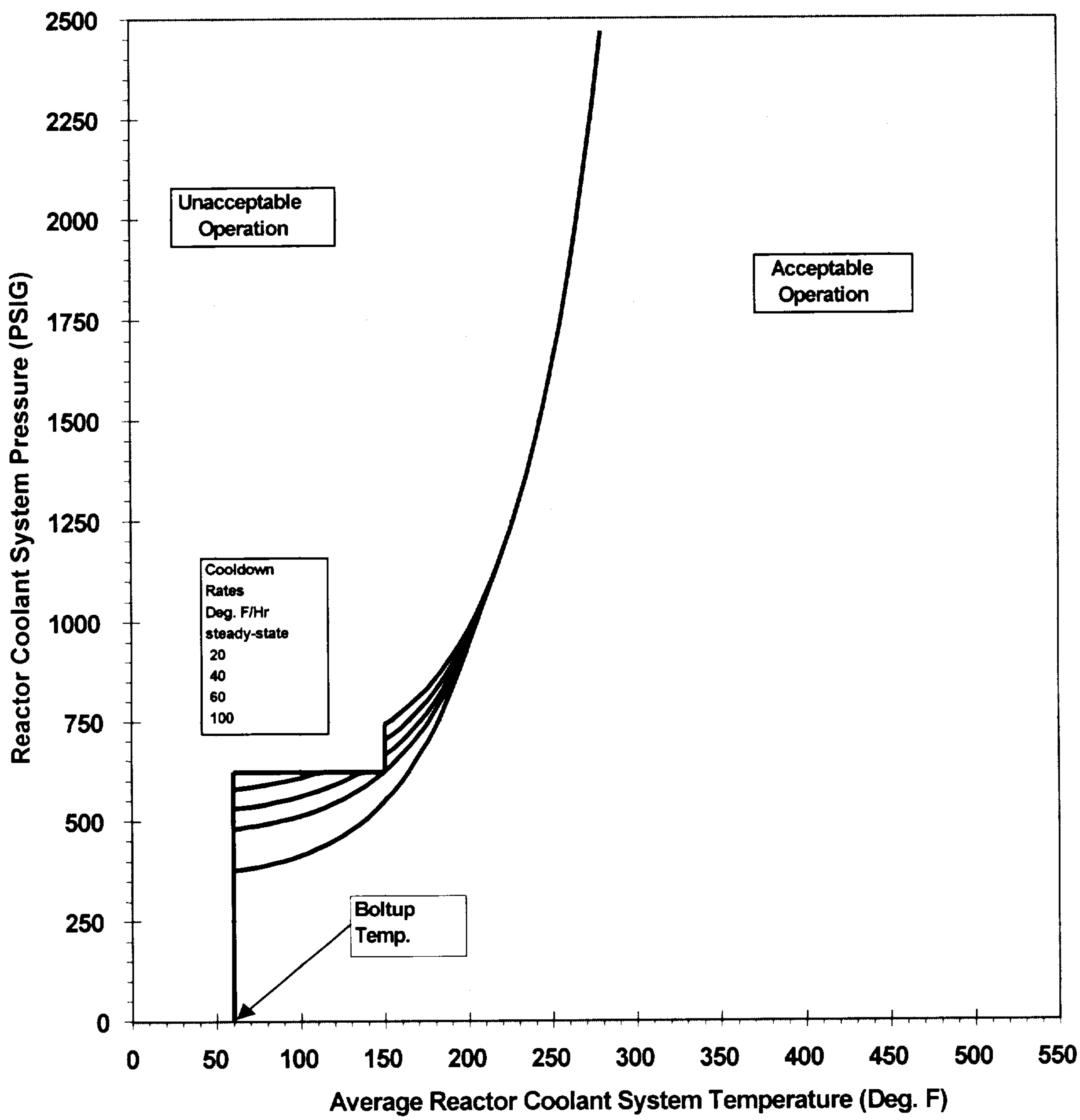


FIGURE 3.4-3

REACTOR COOLANT SYSTEM, PRESSURE - TEMPERATURE, LIMITS FOR VARIOUS COOLDOWN RATES

Insert 2

Reactor Coolant System Cooldown Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F



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 4.1.4 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.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of ~~15~~32 EFPY.

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 must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

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 ~~15~~32 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The ~~15~~32 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515, Revision 1, report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated ~~February 1993~~May 2002.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

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

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown 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.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152° F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stem generator less than or equal to 50°F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs 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. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

Attachment 2 to AEP:NRC:2349-01

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES
UNIT 2

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Reactor Coolant System Heatup Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F

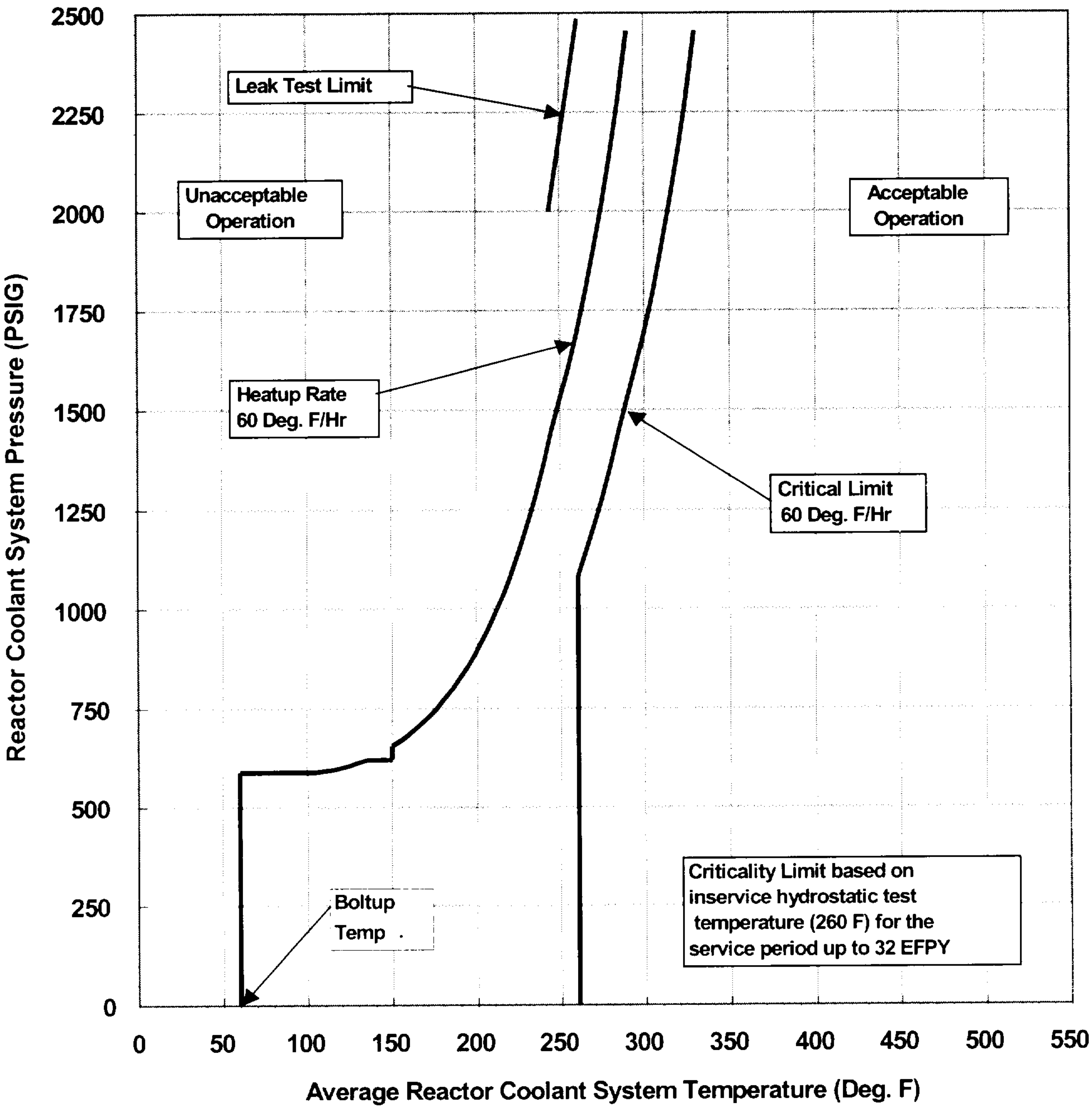

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR $60\,^{O}$F/HR RATE, CRITICALITY LIMIT, BOLTUP LIMIT, AND LEAK TEST LIMIT

Reactor Coolant System Cooldown Limitations Without Margins for Instrumentation Error
Applicable for 32 EFPY of Operation
Limiting Material: Intermediate Shell Plate C5556-2, Cu = 0.15%, Ni = 0.57%
Initial ART: 58 Deg. F, Limiting ART Values at 32 EFPY: 1/4T = 200 Deg. F, 3/4T = 169 Deg. F

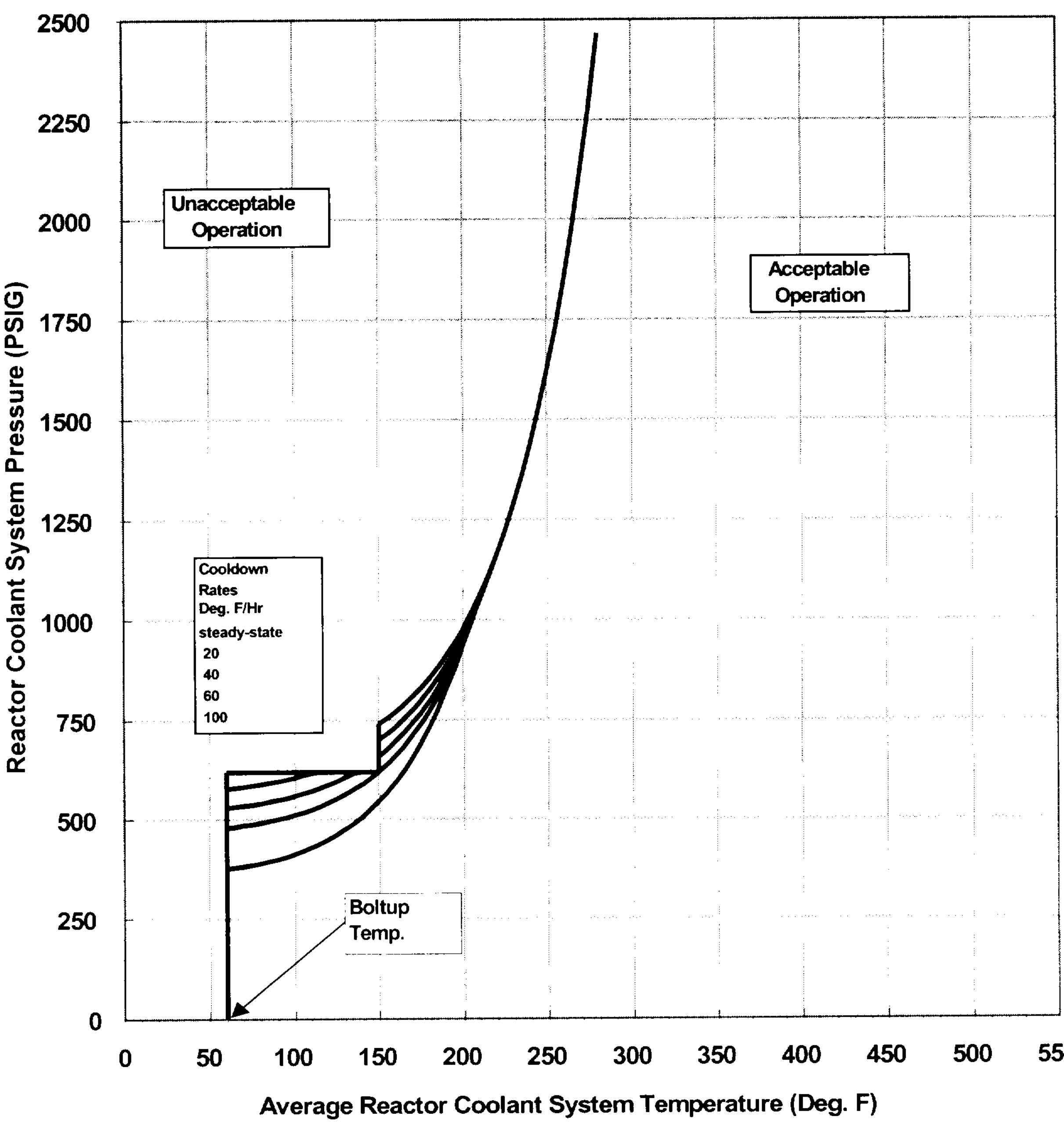

FIGURE 3.4-3
REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE, LIMITS FOR VARIOUS COOLDOWN RATES

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 4.1.4 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.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60 F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of 32 EFPY.

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 must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

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 32 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 32 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515, Revision 1, report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated May 2002.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

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

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown 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.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152 °F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stem generator less than or equal to 50° F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs 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. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

Attachment 3 to AEP:NRC:2349-01

WCAP-13515, Revision 1
"Analysis of Capsule U from Indiana Michigan Power Company D. C. Cook Unit 2
Reactor Vessel Radiation Surveillance Program"
Dated April 2002