ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 7, 1999

Proposed Amendment 160

Justification for Proposed Exemption Request
From Requirements of 10 CFR 50.60, 10 CFR 50.61
and
Appendices G and H to Part 50

Justification For Exemption From the Requiremeuts of 10 CFR 50.60, 10CFR50.61, aud Appendices G aud H to Part 50

The intent of 10 CFR 50.60 and 10 CFR 50.61 is to establish criteria that ensures that each reactor vessel has adequate fracture toughness. When these rules were first promulgated fracture toughness specimens were too large to be used in reactor vessel radiation surveillance capsule programs. Therefore, smaller Charpy V-notch specimens were used to estimate and monitor fracture toughness instead.

The use of Charpy V-notch specimens resulted in the current use of RT_{NDT} as an indexing temperature for the K_{IC} curve employed in Appendix G of 10 CFR 50 and ASME B&PVC Section XI. The development of the current RT_{NDT}-based methodology resulted in both implicit and defined margins. The use of Charpy V-notch specimens to estimate the fracture toughness of a reactor vessel material could not be accomplished without introducing some uncertainty. Therefore, it was considered appropriate to include implicit margins at the time. Implicit margins on fracture toughness derived from nil-ductility temperature testing (as per ASTM E208) and from Charpy V-notch testing (as per ASTM E23) vary considerably from reactor to reactor, i.e., from material to material, for the current RT_{NDT}-based methodology. In view of this implicit margin variability, the only implicit margin that can be considered as required and consistent with the current licensing basis is the margin associated with the most limiting material used to establish the current ASME K_{IC} curve. Any implicit margin exceeding this minimum level has, by definition, never been a requirement for all commercial nuclear power reactors operating in the United States. The material having the minimum implicit margin is a plate of A533B Class 1 steel designated "HSST-02." This plate is one of several correlation monitoring materials used as a basis for comparison between surveillance capsule results from various commercial nuclear power reactor vessels. The methodology proposed by ASME Code Case N-629 and detailed in WCAP-15075 for direct measurement of irradiated fracture toughness, ensures that implicit margins are slightly increased relative to this historically accepted minimum. This proposed methodology also permits comparisons to be made between reactor vessels in terms of both the implicit and defined margins and with respect to the PTS screening criteria.

This document summarizes the basis for the exemption request to use ASME Code Case N-629, ASTM E185-98, ASTM E-1921-97, and the methodology described under Case #6 in WCAP-15075 for establishing end of life and end of life extension indexing reference temperature values for assessment of the integrity of the Kewaunee reactor vessel. A description of the exemptions requested and proposed alternatives is shown below:

Description	Existiug Requiremeut	Proposed Alternative
Determination of adjusted/indexing reference temperatures	10 CFR 50.61	ASME Code Case N-629 and WCAP-15075

Description	Existing Requirement	Proposed Alternative
Use of the latest edition of ASTM E185-98	App H to Part 50 specifies use of ASTM E185-73,-79, -82 for testing of surveillance materials	ASTM E185-98 allows use of ASTM E1921-97 for testing of surveillance capsule material
Use of alternative testing methods for determination of fracture toughness	Appendices G and H to Part 50 specifies Charpy V-Notch impact and drop weight testing	ASTM E1921-97

10 CFR 50.12(a) states that the NRC may grant exemptions from the requirements of the regulations contained in 10 CFR 50 provided that:

(1) the exemption is authorized by law;
(2) the exemption does not present an undue risk to the public health and safety;
(3) the exemption is consistent with the common defense and security; and
(4) special circumstances, as defined by 10 CFR 50.12(a)(2), are present.

This exemption request meets the criteria set in 10CFR50.12 as discussed herein. Additional technical bases for the proposed exemptions are provided as attachments to Reference 1.

1. <u>The requested exemption is authorized by law.</u>

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. <u>The requested exemption does not present an undue risk to the public health and safety</u>.

   10 CFR 50 Appendices G and H specify that surveillance capsules shall be tested in accordance with ASTM E 185-73, -79, and -82. The latest version of ASTM E-185-98 encourages that supplemental fracture toughness testing be conducted in accordance with procedures and requirements of Practice E636, Method E-1820, or Method E-1921 when the surveillance materials exhibit marginal properties. Fracture toughness testing of reconstituted KNPP and Maine Yankee surveillance capsule 1P3571 weld metal has been performed to satisfy the requirements established in accordance with ASTM E-1921. The reconstituted surveillance capsule specimens have been irradiated to end of life and beyond end of life extension fluence. The proposed approach for establishing end of life and end of life extension indexing reference temperature values use fracture toughness results obtained from reconstituted surveillance capsule specimens for both the KNPP and Maine Yankee radiation surveillance programs. Margins are included in the approach that are consistent with those established in 10 CFR 50.61 and 10 CFR 50 Appendix G.

The fracture toughness for the Kewaunee vessel, based on end of life and end of life extension indexing reference temperature values, has been used for the PTS evaluation. With use of Code Case N-588 and Code Case N-629, the weld metal fracture toughness results from the Maine Yankee radiation surveillance program, and the methodology described under Case #6 in WCAP-15075, the reactor vessel intermediate forging and head flange become the controlling and limiting materials for development of the heatup and cooldown limit curves. Since actual fracture toughness measurements now exist for weld metal 1P3571, operating limits can now be established with much more certainty. Using this approach, appropriate margins defined in 10 CFR 50.60 and 10 CFR 50.61 are maintained and unnecessary plant operational restrictions are avoided. The use of this proposed approach ensures that the intent of the requirements specified in 10 CFR 50.60 and 10 CFR 50.61 are satisfied. Therefore, this exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security.

The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60.

Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption request meets the special circumstances of paragraphs:

(a)(2)(ii), Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or it is not necessary to achieve the underlying purpose of the rule; or
(a)(2)(iii), Compliance would result in undue hardship or other cost that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

10 CFR 50.12(a)(2)(ii)

The intent of 10 CFR 50.60 and 10 CFR 50.61 is to establish criteria that ensures that each reactor vessel has adequate fracture toughness. Westinghouse Electric Corporation and ATI Consulting have prepared five WCAP reports to assess and document the integrity of the KNPP reactor vessel relative to the requirements and intent of 10 CFR 50.60, 10 CFR 50.61, Appendices G and H to 10 CFR Part 50. The following five WCAP reports were provided to the NRC as attachments to Reference 1. These provide the technical justification for the exemption request, PTS evaluation, and proposed amendment.

Identification	Title	Description
WCAP-14279 Revision 1	Evaluation of Capsules from the Kewaunee and Capsule A-35 from the Maine Yankee Nuclear Plant Reactor Vessel Radiation Surveillance Programs	Documents standard and supplemental surveillance capsule fracture toughness testing results for the 1P3571 weld metal both in the unirradiated and irradiated condition
WCAP-15075	Master Curve Strategies for RPV Assessment	Methodology to determine EOL and EOLE extension reference temperatures based on the Master Curve Approach
WCAP-14280 Revision 1	Evaluation of Pressurized Thermal Shock for the Kewaunee Reactor Vessel	Includes various Pressurized Thermal Shock (PTS) evaluations for KNPP conducted in accordance with the methodology given in 10 CFR 50.61 and the Master Curve Approach
WCAP-14278 Revision 1	Kewaunee Heatup and Cooldown Limit Curves for Normal Operation	Heatup and cooldown limit curves corresponding to end of plant life fluence
WCAP-15074	Evaluation of the 1P3571 Weld Metal from the Surveillance Programs for Kewaunee and Maine Yankee	Summary of the KNPP 1P3571 weld metal surveillance capsule test results performed to date

Background for much of this work is linked to ongoing efforts by the NRC staff to generically resolve concerns raised during their review of reactor vessel integrity for the Yankee Nuclear Power Station. As part of this effort, the NRC staff issued Generic Letter 92-01, Revision 1 and Generic Letter 92-01, Revision 1, Supplement 1. These generic communiqué obtained information that permitted the NRC staff to independently assess and ensure that licensees are in compliance with the requirements regarding reactor pressure vessel integrity. During review of the industry responses to Generic Letter 92-1, Revision 1 and Generic Letter 92-01, Revision 1, Supplement 1 the NRC Staff discovered inconsistencies within the industry concerning the methodology used to assess reactor pressure vessel integrity including:

1. Large variability in the reported chemistries, i.e., copper and nickel contents, for welds fabricated from the same heat of weld wire.
2. Different initial properties (RT_{NDT}) for welds fabricated from the same heat and weld wire.

3. Different transition temperature shifts for welds fabricated from the same heat and weld wire.
4. Operation with irradiation temperature less than 525°F.
5. Different approaches for determining fluence of the limiting material.

In response to these findings and to provide assurance that all plants will maintain adequate protection against PTS events, the NRC staff required PTS evaluations be performed using conservative inputs. Applying too much conservatism can create the illusion that a reactor vessel is unsafe to operate when in fact it possesses sufficient fracture toughness. Furthermore, if too much conservatism is applied, the overall affect can be a decrease in plant safety because of unnecessary changes made to plant operations and design for the sole reason of addressing an overly conservative PTS evaluation.

Furthermore, the increase in conservatism imposed on the reactor vessel integrity process predicts that the KNPP reactor vessel will reach the PTS screening criteria at an earlier date than that given by the PTS methodology given in 10 CFR 50.61.

10 CFR 50.60 and 10 CFR 50.61 establish criteria that ensures reactor vessels have adequate fracture toughness. The rules preclude fracture during any condition of normal operation, including anticipated operational occurrences, and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. When these rules were first promulgated, fracture toughness specimens were too large to be used in reactor vessel radiation surveillance capsule programs. Therefore, smaller Charpy V-notch specimens were used to estimate and monitor the fracture toughness. The use of smaller Charpy V-notch specimens resulted in the current use of RT_{NDT} as an indexing temperature for the K_{IC} curve employed in Appendix G of 10 CFR 50 and ASME B&PVC Section XI. The development of the current RT_{NDT}-based methodology resulted in both implicit and defined margins. Use of Charpy V-notch specimens to estimate the fracture toughness of a reactor vessel material could not be accomplished without some uncertainty. Due to this uncertainty, it was considered appropriate at that time that these implicit margins exist. However, the implicit margins on fracture toughness derived from nil-ductility temperature testing (as per ASTM E208) and from Charpy V-notch testing (as per ASTM E23) vary considerably from reactor to reactor, i.e., from material to material, for the current RT_{NDT}-based methodology. In view of this implicit margin variability, the only implicit margin that can be considered as required and consistent with the current licensing basis is the margin associated with the most limiting material used to establish the current ASME K_{IC} curve. Any implicit margin exceeding this minimum level has, by definition, never been a requirement for all commercial nuclear power reactors operating in the United States. The material having the minimum implicit margin is a plate of A533B Class 1 steel designated "HSST-02". The methodology proposed by ASME Code Case N-629, and detailed in WCAP-15075, ensures that implicit margins are somewhat increased relative to this historically accepted minimum. The proposed methodology also permits comparisons to be made between reactor vessels in terms of both the implicit and defined margins and with respect to the PTS screening criteria.

10 CFR 50 Appendix G specifies that surveillance capsules shall be tested in accordance with ASTM E 185-73, -79, -82. The latest version of ASTM E 185-98 recommends that supplemental fracture toughness testing be conducted in accordance with procedures and requirements of Practice E636, Method E 1820, or Method E 1921 when the surveillance materials exhibit marginal properties. Fracture toughness testing of reconstituted KNPP and Maine Yankee surveillance capsule 1P3571 weld metal has been performed to satisfy the requirements established in accordance with ASTM E-1921. The reconstituted surveillance capsule specimens have been irradiated to EOL and beyond EOL extension fluence. End of life and end of life extension indexing reference temperature values were then determined using fracture toughness data obtained from these tests in accordance with ASTM E1921-97, implicit margins, and defined margins that are consistent with those established in Regulatory Guide 1.99 Revision 2 and 10 CFR 50 Appendix G.

T_0 indexes the variation of fracture toughness with temperature. The ASME Code recently adopted a T_0-based index temperature (termed RT_{T0}) as an alternative to RT_{NDT} for indexing of the ASME K_{IC} curve. ASME Code Case N-626 endorses the following relationship:

$$RT_{To} = T_0 + \Delta°F.$$

with a value of 35°F for the Δ term. To maintain linkage with historically accepted safety margins, Δ is selected such that the K_{IC} curve, when indexed to RT_{To}, bounds available fracture toughness data in a manner functionally equivalent to how the K_{IC} curve indexed to RT_{NDT} bounds available fracture toughness data. Δ has been defined to ensure that the data set which established the position of the K_{IC} curve when indexed to RT_{NDT} (HSST-02) has an equal or greater implicit margin (i.e., proximity of toughness data to the bounding curve) when the K_{IC} curve is indexed to RT_{To}. The level of margin implicit to the current assessment methodology (i.e., a RT_{NDT} indexed K_{IC} curve) for the most limiting material, HSST-02, is 17°F. ASME Code Case N-626 adopted a Δ value of 35°F, a value used for the Kewaunee application. The adoption of a Δ value of 35°F exceeds the value needed to maintain consistency with the current licensing basis by 18°F. Additionally, a measurement uncertainty on RT_{To} of two standard deviations ($2\sigma_{To}$) is also used for this application. For the KNPP application the measurement uncertainty has been shown to be $2\sigma_{To}$=16°F. Finally, the material heat uncertainty is accounted for in a manner consistent with the ratio technique prescribed in Regulatory Guide 1.99 Rev. 2. The material heat uncertainty based upon fracture toughness testing results of reconstituted irradiated surveillance capsule specimens from KNPP and Maine Yankee has been shown to be 35°F. Thus, several margins have been explicitly included to the measured RT_{To} value for KNPP:

1. 16°F, a $2\sigma_{To}$
2. 17°F, a margin that increases the spacing of fracture toughness data relative to the bounding K_{IC} curve in excess of that which has been historically accepted (and required)
3. 35°F, a material heat uncertainty

It should be noted that the sum of the first two margins, a total of 33°F, are not required by the current regulations but are added to be consistent with good engineering practice (for margin #1), and to be consistent with current ASME guidelines (for margin #2).

In summary, fracture toughness based end of life and end of life extension indexing reference temperature values will be used for the PTS evaluation of the KNPP reactor vessel. With NRC approval to use Code Case N-588, Code Case N-629, the weld metal fracture toughness results, and the methodology given in WCAP-15075, the reactor vessel intermediate forging and head flange material will become the limiting and controlling materials for development of the heatup and cooldown limit curves. The revised heatup and cooldown limit curves have been developed using the mechanical properties associated with the intermediate forging and methodology given in 10 CFR 50.61 and 10 CFR 50, Appendix G. Since actual fracture toughness measurements now exist for weld metal 1P3571 operating limits can now be established with much more certainty. This ensures appropriate margins consistent with those defined in 10 CFR 50.60 and 10 CFR50.61 are maintained and unnecessary plant operational restrictions are avoided.

Therefore, determining end of life and end of life extension indexing reference temperature values by using ASME Code Case N-629, ASTM E185-98, ASTM E1921-97, the methodology described under Case #6 in WCAP-15075, and measuring the fracture toughness from reconstituted surveillance capsule specimens obtained from the KNPP and Maine Yankee reactor vessel surveillance programs, satisfies the underlying purpose of the NRC regulations to ensure an acceptable level of safety.

10 CFR 50.12(a)(2)(iii)

The use of conservative inputs in combination with both the implicit and defined margins produces a wide range of results for evaluation of PTS. Four cases, using Chary V-notch data from heat 1P3571 weld metal, are evaluated and documented in WCAP-14280 Rev 1 (Attachment 5 of Reference 1) and a fifth scenario is discussed. The resultant PTS values range from 267°F to 292°F for these four cases. Conversely, the resultant PTS value is well in excess of the PTS screening criteria if the Maine Yankee initial properties and chemistry factor are combined with the KNPP EOL fluence.

If Maine Yankee initial properties and chemistry factor inputs in combination with the KNPP EOL and EOL extension fluences are considered, as they have been required to be in the past, Wisconsin Public Service Corporation and its customers would experience undue hardship and expense without any increase in safety. Even the possibility of having to include additional margin to a PTS value of 292°F to account for irradiation below 540°F, or variation in copper in excess of that assumed when 10 CFR 50.61 was written would produce undue hardship and costs to perform engineering evaluations or a thermal annealing treatment of the reactor vessel. Furthermore, changes made to operating procedures such as restrictions placed on operation of reactor coolant pumps or a higher LTOP enabling temperature may result in a decrease in overall plant safety due to a smaller operating window and the possibility of inadvertently lifting the LTOP safety valve which would subject the reactor vessel to a sudden pressure change.

KNPP started commercial operation in June of 1974. The plant is licensed to operate through 2013. Evaluation of the integrity of the KNPP reactor vessel has been a priority for the KNPP staff. WPSC is planning to assess opportunities for operation of the Kewaunee plant beyond 2013. Operation beyond year 2013 is strongly contingent upon the acceptability of establishing end of life and end of life extension indexing reference temperature values by using ASME Code N-629, ASTM E185-98, ASTM E1921-97, the methodology described under Case #6 in WCAP-15075, and measuring the fracture toughness from reconstituted surveillance capsule specimens obtained from the KNPP and Maine Yankee reactor vessel surveillance programs. The discussion of special circumstances under 10 CFR 50.12(a)(ii) above clearly shows that the proposed methodology is acceptable, and that application of 10 CFR 50.60 and 10 CFR 50.61 for establishing end of life and end of life extension indexing reference temperature values from Charpy V-notch and drop weight testing is not necessary to achieve the underlying purpose of the rule. Therefore, continued compliance with the rule would result in undue hardship and costs to Wisconsin Public Service Corporation and its customers.

ATTACHMENT 2

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 7, 1999

Proposed Amendment 160

Background
Description of Proposed Changes
Safety Evaluation
Significant Hazards Determination
Environmental Considerations

Proposed Amendment No. 160 to the KNPP Technical Specifications

BACKGROUND

This proposed amendment (PA) is being submitted to revise the current PTS evaluation, LTOP limitations, and heatup and cooldown limit curves for the Kewaunee Nuclear Power Plant (KNPP). As discussed in Reference 2, the current curves and LTOP limit are restricted to 28 EFPY. The purpose of this amendment request is to revise the PTS evaluation and obtain limit curves and LTOP limits that are valid through 33 EFPY.

The PTS evaluation is being revised at this time to resolve several long standing issues in order to demonstrate that it is safe to operate the KNPP reactor vessel through end of life extension. These reactor vessel integrity assessments satisfy the requirements of 10 CFR 50.60 and 10 CFR 50.61 except that two(2) proposed alternatives to these requirements have been incorporated into the reactor vessel integrity evaluations.

In reference 1, WPSC requested permission under 10 CFR 50.12 to use ASME B&PVC Case N-588. The NRC authorized this exemption request in References 4 and 5.

Attachment 1 of this submittal provides justification and requests permission under 10 CFR 50.12 to determine end of life (EOL) and EOL extension indexing reference temperature values by using ASME Code Case N-629, ASTM E185-98, ASTM E1921-97, the methodology described in case #6 of WCAP-15075, and measuring fracture toughness from reconstituted surveillance capsule specimens obtained from the Kewaunee and Maine Yankee reactor vessels. The proposed alternatives described in the requested exemptions provides a more accurate and yet conservative assessment of PTS for the KNPP reactor vessel.

With NRC approval of the proposed exemption in attachment 1 and Code Case N-588, the reactor vessel intermediate forging and head flange become the limiting and controlling materials for development of the heatup and cooldown limit curves. The revised heatup and cooldown limit curves have been developed using the mechanical properties associated with the reactor vessel intermediate forging and head flange and methodology given in 10 CFR 50.61 and 10 CFR 50, Appendix G. The corresponding LTOP system enabling temperature is also based upon the mechanical properties associated with the reactor vessel intermediate forging and head flange, Code Case N-514, and methodology given in 10 CFR 50.61 and 10 CFR 50, Appendix G.

The KNPP Technical Specifications (TS) require that reactor coolant temperature, pressure and system heatup and cooldown rates be limited in accordance with the heatup and cooldown curves and corresponding LTOP enabling temperature. The current limit curves and LTOP limits were originally docketed on November 18, 1998, under PA 157, and are intended to be applicable through 33 effective full power years (EFPY). However, due to a technical disagreement between WPSC and NRC (see Reference 2), the current limit curves and LTOP limits have been limited to 28 EFPY.

The heatup and cooldown curves proposed by this amendment are valid up to 33 EFPY. They are based on Charpy V-notch and drop weight data representative of the reactor vessel head flange and intermediate forging material. Calculational methods used are prescribed in 10 CFR 50.61; ASME Code Case N-588; ASME Code Case N-629; ASTM E185-98; ASTM E 1921-97; Appendix G to the ASME Code; and Chapter 5.3.2 of NRC Regulatory Standard Review Plan. Inherent conservatism imposed by these requirements include:

- using a safety factor of 2 on the principal membrane (pressure) stresses,
- assuming a flaw at the surface with a depth one-quarter (1/4) of the vessel wall thickness and a length of six (6) times its depth,
- using a conservative fracture toughness curve that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the Kewaunee reactor vessel material, and
- adding margin to cover uncertainties in the EOL and EOLE values of RT_{PTS}/RT_{NDT}, fluence and the calculational procedures.

The basis for the proposed curves is contained in WCAP-14278 Revision 1, titled, "Kewaunee Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation" which is provided in Attachment 6 of Reference 1.

Regulatory Requirements for Fracture Toughness and Pressure - Temperature (P/T) Limitations

10 CFR 50.60(a) states that the reactor coolant pressure boundary must meet the fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50. 10 CFR 50.60(b) states that proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under 50.12. In References 4 and 5, the NRC granted WPSC approval to use ASME Code Case N-588 for development of Appendix G pressure and temperature limitations. This proposed TS amendment is further contingent upon NRC approval of the exemption request described in attachment 1 to this letter. With NRC approval, the reactor vessel intermediate forging and closure head flange become the limiting and controlling materials for derivation of the revised heatup and cooldown limit curves and corresponding LTOP system enabling temperature. The regulatory requirements for heatup and cooldown and normal low temperature overpressure protection are as follows:

1) 10 CFR 50, Appendix G.IV.A: Pressure retaining components of the reactor coolant pressure boundary must meet the requirements of the ASME Code, supplemented as follows for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. For the reactor vessel beltline materials, including welds, plates, and forgings, the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of 10 CFR Part 50. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

2) 10 CFR 50, Appendix G.IV.1.a: Reactor vessel beltline materials must have Charpy upper-shelf energy, in the transverse direction for the base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code incorporated into 10 CFR 50.55a(b)(2) at the time the analysis is submitted.

3) 10 CFR 50, Appendix G.IV.A.2.a: Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 3 (of Appendix G), and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical.

4) 10 CFR 50, Appendix G.IV.A.2.b: The pressure-temperature limits identified as "ASME Appendix G limits" in Table 3 (of Appendix G) require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

5) 10 CFR 50, Appendix G.IV.A.2.c: The minimum temperature requirements given in Table 3 (of Appendix G) pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 3 (of Appendix G), the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical.

10 CFR 50.61 identifies the general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. 10 CFR 50.61 describes two methods acceptable to the NRC staff to evaluate the predictions of radiation embrittlement needed to implement Appendices G and H to 10 CFR Part 50. Paragraph (c)(2)(ii)(A) of 10 CFR 50.61 requires that licensees determine a material-specific value of chemistry factor when the surveillance data is deemed credible according to the criteria of paragraph (c)(2)(I) of 10 CFR 50.61. If the surveillance data is not deemed credible then the chemistry factor is determined using the copper and nickel content according to the criteria of paragraph (c)(1)(iv)(A) of 10 CFR 50.61. Additionally, paragraph (c)(2)(ii)(B) specifies if the chemical content of the surveillance weld differs from the average for the weld wire heat number associated with the vessel weld, then the measured values of ΔRT_{NDT} be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that of the surveillance weld.

The intermediate forging and circumferential weld are located in the beltline region of the KNPP reactor vessel. The proposed heatup and cooldown limit curves have been constructed by combining the most conservative pressure temperature limits of these limiting materials: intermediate forging, beltline weld, and closure head flange. The revised heatup and cooldown limit curves and LTOP system enabling temperature are based upon RT_{NDT}/RT_{PTS} values of the intermediate forging and closure head flange materials which are more restrictive than the fracture toughness indexing reference temperature values obtained for the beltline circumferencing weld.

The surveillance capsule data for the intermediate forging does not meet one of the credibility criteria, thus the chemistry factor is given by Table 1 of 10 CFR 50.61 as a function of copper and nickel content. There is no surveillance capsule data for the closure head flange, thus the chemistry factor is taken from Table 1 of 10 CFR 50.61. For purposes of developing the heatup and cooldown limit curves, WPSC has used the copper and nickel content for the intermediate forging for determining the chemistry factor of the forging.

DESCRIPTION OF PROPOSED CHANGES TO SECTION TS 3.1, AND FIGURES TS 3.1-1 AND TS 3.1-2

PA 160 extends application of the Appendix G heatup and cooldown limits from 28 EFPY to 33 EFPY. With NRC approval to use ASME Code Cases N-588 and the exemption requested in attachment 1, the allowable combination of pressure-temperature for the cooldown limits is marginally greater than the current limits. This TS amendment does not affect the current LTOP setpoint, LTOP system enabling temperature, or allowable combination of pressure-temperature for the heatup limits. However, this TS amendment does extend the applicability date for the current LTOP setpoint, LTOP system enabling temperature, and allowable combination of pressure-temperature for the heatup limits from 28 EFPY to 33 EFPY. The LTOP setpoint, LTOP system enabling temperature, and allowable combination of pressure-temperature for the heatup limits in the current TS's were limited to 28 EFPY by Reference 2. The technical content of this proposed amendment is identical to PA 157 for the LTOP setpoint, LTOP system enabling temperature, and allowable combination of pressure-temperature for the heatup limits.

This proposed change will modify the limiting conditions for operation for reactor coolant temperature and pressure as summarized below:

1) TS 3.1.b.1 is modified to reflect applicability of Figure TS 3.1-1 and Figure TS 3.1-2 for the service period up to 33 effective full-power years. This makes TS 3.1.b.1 consistent with the expiration date noted on the revised heatup and cooldown limit curves.

2) Figures TS3.1-1 and TS 3.1-2 have been modified to define 10 CFR 50 Appendix G pressure temperature limitations applicable for the service period of up to 33 effective full power years.

3) Basis Section 3.1 has been revised accordingly.

4) The List of Figures has been modified to reflect a new expiration date of 33 effective full power years for Figure TS 3.1-1 and Figure 3.1-2.

The temperature restriction cited in TS 3.1.a.1.C, TS 3.1.b.1.C, and TS 3.1.b.4 is commonly referred as the low temperature overpressure protection (LTOP) system enabling temperature. In accordance with Branch Technical Position RSB 5-2, "Overpressurization Protection of PWRs While Operating at Low Temperatures," which is attached to Standard Review Plan Section 5.2.2, this parameter has historically been defined as RT_{NDT} + 90°F. The KNPP temperature restriction is based on a new definition for enabling temperature which is defined (RT_{NDT} + 50°F) in ASME B&PVC Section XI Code Case N-514. The new temperature restriction value is calculated using Code Case N-514, and the end of life (EOL) reference temperature (RT_{PTS}) and adjusted reference temperature (RT_{NDT}) values associated with the intermediate forging. This combination of inputs results in more limiting pressures and temperatures in the low temperature region of the isothermal cooldown limit curve, where LTOP events are a concern, than the parameters associated with the beltline weld when Code Cases N-588 and N-629 are applied. ASME Code Case N-514 was accepted for use at KNPP by the NRC in Reference 6.

Calculation of Appendix G Pressure-Temperature Limits

Maximum allowable pressures, reactor coolant temperature, and system heatup and cooldown rates have been determined for the KNPP reactor coolant system. The limiting and controlling materials are the vessel closure head flange, beltline weld, and intermediate shell forging. The limits ensure integrity for hydrostatic pressure and leak tests, normal operation including anticipated operational occurrences, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., ≤ 200°F)). These limitations have been calculated using the approved methodology described in 10 CFR 50.61, the 1989 Edition of Appendix G of Section XI of the ASME Code, ASME Code Case N-514 , ASME Code Case N-588, and ASME Code Case N-629 as an alternative to 10 CFR 50.61. The calculation used:

1. The neutron fluence (E > 1 MEV) values based on projected operating hours through the end of the life (12-31-13) and end of the life extension period (12-31-33). EOL and EOLE correspond to 33 EFPY and 51 EFPY, respectively. The existing heatup and cooldown limit curves are based on neutron fluence values thru 33 EFPY. The current LTOP enabling temperature is also based on EOL fluence projections (33 EFPY). The actual expiration date of the current operating license is December 21, 2013. Both the existing and proposed enabling temperature and limit curves have been derived using the latest ENDF-VI dosimetry cross sections. The Kewaunee radiation surveillance program consists of six (6) capsules: V, R, P, T, S, and N. Four (4) capsules (V,R,P, and S) monitor the reactor vessel through EOL fluence. Capsule N will monitor the reactor vessel for the life extension period and Capsule T is an extra standby capsule. The revised heatup and cooldown limit curves and LTOP system enabling temperature are based on radiation surveillance information from capsules V, R, P, and S. Capsule S was removed from the KNPP reactor vessel in 1995. The current heatup and cooldown limit curves were incorporated into the TS's in 1999 and used the same neutron

fluence values to derive the proposed enabling temperature and heatup and cooldown limit curves.

2. Paragraph (c)(1)(iv)(A) of 10 CFR 50.61. Paragraph (c)(1)(iv)(A) of 10 CFR 50.61 requires that licensees determine a chemistry factor using the best estimate Wt% copper and Wt% nickel. This paragraph is applicable for establishing the chemistry factor, since the surveillance data for the intermediate forging does not satisfy the credibility criteria. For the intermediate forging, which is controlling for the proposed heatup and cooldown limit curves, the chemistry is 0.06 Wt% Cu and 0.71 Wt% Ni. Table 2 provides a chemistry factor value of 37°F. Adjustment of this chemistry factor is not required, since the test specimens have been obtained from the intermediate forging used to fabricate the KNPP reactor vessel.

3. The lower temperature portion of the current Appendix G cooldown limit curves and heatup limit curves (and corresponding LTOP system enabling temperature) are based on the following inputs and intermediate forging related data: initial RT_{NDT} of 60°F; 0.06 w/o Cu and 0.71 w/o Ni; chemistry factor value of 37°F derived in accordance with paragraph (c)(1)(iv)(A) of 10 CFR 50.61; a margin term of 34°F; and EOL fluence projected through 33 EFPY.

   The higher temperature region of the current Appendix G cooldown limit curves is based on the following inputs and circumferential weld related properties: initial RT_{NDT} of -50°F; an adjusted chemistry factor value of 219.9°F derived in accordance with paragraph (c)(2)(ii)(A) of 10 CFR 50.61; a margin term of 28°F; and EOL fluence projected through 33 EFPY.

   The horizontal and vertical portions of the current cooldown and heatup limit curves, located between 73°F to 273°F, is limited by the closure head flange unirradiated RT_{NDT} of 60°F.

   The proposed Appendix G limit curves (and corresponding LTOP system enabling temperature) are based on the following inputs and intermediate forging related data: initial RT_{NDT} of 60°F; 0.06 w/o Cu and 0.71 w/o Ni; chemistry factor value of 37°F derived in accordance with paragraph (c)(1)(iv)(A) of 10 CFR 50.61; a margin term of 34°F; and EOL fluence projected through 33 EFPY. The horizontal and vertical portions of the proposed cooldown and heatup limit curves, located between 73°F to 273°F, is limited by the closure head flange unirradiated RT_{NDT} of 60°F. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel 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. These limits are incorporated into the pressure-temperature limit curves wherever applicable.

   This proposed amendment ensures that the basis for the revised heatup and cooldown limit curves and revised LTOP system enabling temperature is the same.

4. The proposed LTOP enabling temperature is based on the definition provided in Code Case N-514. The NRC accepted application of this Code Case N-514 in Reference 6. The intent of the LTOP enabling temperature is to protect against exceeding the Appendix G curve during either a mass or energy input transient. The most limiting proposed Appendix G curve is the isothermal cooldown limit curve which is a composite curve representing the intermediate forging and closure head flange properties. The proposed enabling temperature does not apply a chemistry factor ratio, since the applicable revised limit curve is based on the intermediate forging properties.

   The proposed isothermal cooldown limit curve and corresponding LTOP system enabling temperature are based on the following inputs and intermediate forging properties: initial RT_{NDT} of 60°F; 0.06 w/o Cu and 0.71 w/o Ni; chemistry factor value of 37°F derived in accordance with paragraph (c)(1)(iv)(A) of 10 CFR 50.61; a margin term of 34°F; and EOL fluence projected through 33 EFPY.

   The horizontal and vertical portions of the proposed isothermal cooldown limit curve, located between 73°F to 273°F, corresponds to the closure head flange unirradiated RT_{NDT} of 60°F. 10 CFR 50, Appendix G states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure.

   WCAP-14279, Revision 1 (Attachment 3 of Reference 1), WCAP-15075 (Attachment 4 of Reference 1), WCAP-14280, Revision 1 (Attachment 5 of Reference 1), WCAP-14278, Revision 1 (Attachment 6 of Reference 1), and WCAP-15074 (Attachment 7 of Reference 1) provide supporting documentation for the proposed heatup and cooldown limit curves, PTS evaluation, and LTOP enabling temperature. The proposed heatup and cooldown limit curves along with the corresponding technical specification changes are provided in Attachment 3.

SAFETY EVALUATION

The revised heatup and cooldown limit curves and LTOP enabling temperature have been developed to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. The method used in preparing the heatup and cooldown limit curves is presented in Attachment 6 of Reference 1. Supporting information can be found in Attachment 1 to this letter and in Attachments 3, 4, 5, and 7 of Reference 1. The method used for evaluation of the beltline weld metal and intermediate forging is consistent with ASME Boiler and Pressure Vessel Code Sections III and XI, Appendix G, NRC Regulatory Standard Review Plan Chapter 5.3.2, and 10 CFR 50.61.

The safety factors and margins applied in the preparation of the limit curves and LTOP enabling temperature meet the criteria set forth by these documents.

Inherent conservatism in the P/T limits resulting from these documents include:

a. An assumed defect in the reactor vessel wall with a depth equal to 1/4 of the thickness of the vessel wall (1/4T) and a length equal to 1-1/2 times the thickness of the vessel wall.

b. An assumed reference flaw oriented in both longitudinal and circumferential directions applied to the intermediate forging. With application of Code Case N-588, an assumed reference flaw oriented in the circumferential direction is imposed on the circumferentially oriented beltline weld. At KNPP, the only weld in the core region is oriented in the circumferential direction.

c. A factor of safety of 2 is applied to the membrane stress intensity factor.

d. The limiting toughness is based upon a reference value (K_{IR}) which is a lower bound on the dynamic crack initiation or arrest toughness.

e. A margin of 58 psi and 13°F to account for plant instrumentation uncertainty.

f. A pressure margin of 70 psi for the difference between the instrument and beltline region. Note, WCAP-14278 Revision 1 did not incorporate the pressure difference between the instrument and beltline region into either the Appendix G limits or the tables that summarize the allowable pressure and temperature. A value of 70 psi was subtracted from each of the values listed in WCAP-14278 Revision 1 to generate Figure TS 3.1-1 and Figure TS 3.1-2. The pressure and temperature values for the Leak Test Limit and temperature value for the vertical portion of the criticality limit curve were generated by Westinghouse Electric Corporation and are documented in Reference 7.

g. A 2-sigma margin term of 34°F is applied in determining the adjusted reference temperatures for the intermediate forging.

Since the LTOP enabling temperature is based upon the lower portion of the proposed Appendix G isothermal cooldown limit curve, which is based on the beltline intermediate forging properties, the above margins equally apply to the LTOP enabling temperature. Beyond the conservatism described above, WPSC has incorporated the following additional margin into heatup and cooldown limit curves and LTOP enabling temperature prepared under this PA:

1. Instrumentation margin of 29 psig is included in the LTOP set point determination.

2. Use of very conservative heat transfer coefficients (7000 Btu/ hr-ft^2-°F), and neglecting the effects of cladding conductivity in the analysis of thermal stress in the P/T curves. Further, the thermal stress is calculated for fixed and constant rates of temperature change and does not reflect the intermittent rates actually experienced by the vessel. That is, hold points for items such as crud burst cleanup during shutdown and securing residual heat removal from

service during startup act as a thermal soak period and reduce the integrated effect on thermal stress.

Additional conservatism beyond that described above but not used in development of the proposed TS and Figures include:

a) The assumption of a 1/4 thickness surface flaw. No flaws exceeding the ASME Section XI allowable flaw size for volumetric examination have been detected during either of the two inservice examinations of the reactor vessel performed in 1985 and 1995.

b) A 2 inch diameter spring loaded safety valve set at 480 psig located in the LTOP/RHR system. At 500 psig, the LTOP relief valve setpoint, the relieving capacity of this smaller valve is 230 gpm.

c) The actual LTOP relief valve capacity is at least 10% greater than the capacity used in the design and setpoint analyses. This is in accordance with the requirements of Section III NC-7000.

d) Assumptions in the overpressure transient analyses are conservative relative to the actual Kewaunee reactor coolant system (RCS) and operating practices:

1. The RCS was assumed to be rigid with respect to metal expansion.

2. No credit was taken for the shrinkage effect caused by low temperature safety injection water added to higher temperature reactor coolant.

3. No credit was taken for the reduction in reactor coolant bulk modulus at RCS temperatures above 100°F (constant bulk modulus at all RCS temperatures).

4. The entire volume of water of the steam generator secondary was assumed available for heat transfer to the primary. In reality, the liquid immediately adjacent and above the tube bundle would be the primary source of energy in the transient.

5. The overall steam generator heat transfer coefficient, U, was assumed to be the free convective heat transfer coefficient of the secondary, h_{sec}. The forced convective heat transfer coefficient of the primary, h_{pri} and the tube metal resistance have been ignored thus resulting in a conservative (high) coefficient.

6. The reactor coolant pump start time assumed in the heat input analysis was 9-10 seconds; whereas, the Kewaunee pump startup time is 25-30 seconds.

Alternative methodologies to the safety margins required by Appendix G to 10CFR Part 50 have been developed by the ASME Working Group on Operating Plant Criteria. Three of these methodologies are contained in ASME Code Cases N-514, N-588, and N-629.

Code Case N-514 provides criteria for determining the LTOP enabling temperature. The NRC reviewed and accepted the use of Code Case N-514 for determining the LTOP enabling temperature at KNPP in reference 6. KNPP has utilized Code Case N-514 to establish the LTOP enabling temperature; the provision for exceeding 110% of the Appendix G limits has not been incorporated into this amendment.

Code Case N-588 provides benefits in terms of calculating pressure-temperature limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. NRC reviewed and approved the use of ASME Code Case N-588 in references 4 and 5.

Attachment 1 of this letter provides supporting information justifying application of ASME Code Case N-629.

Four surveillance capsules have been removed and tested from the Kewaunee reactor vessel. The surveillance capsule data has been evaluated to the five credibility requirements of Regulatory Guide 1.99, Revision 2. This evaluation finds that the surveillance date for the circumferential beltline weld metal are credible, however, the surveillance data for the intermediate forging does not satisfy one of the credibility criteria. Thus, Regulatory Position C.1 of Regulatory Guide 1.99, Revision 2 was used for calculating the chemistry factor of the reactor vessel intermediate forging material.

Improvements to a low leakage core design at fuel cycle 16 and all subsequent fuel cycles decreases the rate of shift in transition temperature from ductile to non-ductile behavior.

Compliance with 10 CFR 50.61 and use of ASME Code Cases N-514, N-588, and N-629 is an acceptable approach for evaluating predictions of radiation embrittlement needed to implement Appendices G and H to 10 CFR Part 50. These limits meet the NRC acceptance criteria for the LTOP setpoint and system design as described in NRC Safety Evaluation Report (SER) to WPSC dated September 6, 1985 which concluded that "the spectrum of postulated pressure transients would be mitigated...such that the temperature pressure limits of Appendix G to 10 CFR 50 are maintained."

Radiological off-site exposure from these plant conditions does not exceed the guidelines of 10 CFR 100. Therefore, utilization of the proposed pressure and temperature limit curves will not adversely impact the consequences of any of the accidents in the Kewaunee USAR.

Based on the information above, the preparation of the revised heatup and cooldown limit curves and LTOP system enabling temperature meets the applicable safety criteria and regulatory guidance, and therefore does not represent a safety concern.

SIGNIFICANT HAZARDS DETERMINATION

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Failure of a reactor vessel is not an accident that has been previously evaluated. Design provisions ensure that this is not a credible event. Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Compliance with 10 CFR 50.61, 10 CFR 50 Appendix G and H, and application of ASME Code Case N-514, ASME Code Case N-588, and the exemption requested in attachment 1 ensures that failure of a reactor vessel will not occur. The proposed changes do not impact the capability of the reactor coolant pressure boundary piping (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore do not increase the potential for the occurrence of a LOCA.

   The LTOP setpoint, LTOP system enabling temperature, and revised P/T limits reflected in proposed Figures TS 3.1-1 and TS 3.1-2 ensure that the Appendix G pressure/temperature limits are not exceeded, and therefore, ensure that RCS integrity is maintained. The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, construction standards, or setpoints. The reactor coolant system full power operating pressure (2235 psig) is not being changed by this proposed amendment. The LTOP valve setpoint remains at ≤500 psig. The LTOP enabling temperature based on Figure TS 3.1-2 is 200°F and is consistent with ASME Code Case N-514 guidance of RT_{NDT} + 50°F. The LTOP enabling temperature is not changed by this amendment. The allowable combination of Appendix G pressure and temperature for the cooldown limits is marginally greater than the current limits. The combination of slightly greater allowable Appendix G pressure and temperature limits and low enabling temperature produces an adequate operating window. An adequate operating window reduces the likelihood of inadvertently lifting the LTOP relief valve while maneuvering the plant through the knee of the P-T curve during startup and shutdown. The probability of an LTOP event occurring is independent of the pressure-temperature limits for the RCS pressure boundary and enabling temperature. Therefore, the probability of a LTOP event is not increased.

   The revised heatup and cooldown limit curves and corresponding LTOP enabling temperature were developed using test results from unirradiated and/or irradiated specimens that represent the KNPP reactor vessel beltline circumferential weld, closure head flange, and intermediate forging. The circumferential beltline weld and intermediate forging are the most limiting materials in the reactor coolant pressure boundary. These materials are limiting due to the effects of neutron irradiation which cause the flow properties to increase and the toughness to decrease. The circumferential beltline weld is the controlling material for evaluation of pressurized thermal shock. With NRC approval to use Code Case N-588 and the exemption requested in attachment 1, the reactor vessel intermediate forging and head flange become the limiting and controlling materials for development of the Appendix G limit curves and corresponding LTOP system enabling temperature. 10 CFR

50, Appendix G states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. Fracture toughness, drop weight, and Charpy V-notch testing of the 1P3571 weld metal and drop weight, and Charpy V-notch testing of the intermediate forging material has been performed. The results of those tests have been used for derivation of the revised PTS assessment, the proposed Appendix G heatup and cooldown limit curves, and the corresponding LTOP system enabling temperature. The revised limit curves and corresponding LTOP enabling temperature have been developed using accepted engineering practices. The evaluations were performed in accordance with methods derived from the ASME Boiler and Pressure Vessel Code, criteria set forth in NRC Regulatory Standard Review Plan 5.3.2, and 10 CFR 50.61. The revised heatup and cooldown limit curves and corresponding LTOP enabling temperature ensures adequate fracture toughness for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary. These limit curves provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., ≤ 200°F)), thus ensuring the integrity of the reactor coolant pressure boundary.

The changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the changes do not affect any fission product barrier. The changes do not degrade or prevent the response of the LTOP relief valve or other safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the consequences of an accident previously evaluated will not be increased.

Thus, operation of KNPP in accordance with the PA does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the potential consequences of a reactor vessel failure are so severe, industry and governmental agencies have worked together to ensure that failure will not occur. Application of ASME Code Case N-514, ASME Code Case N-588, and the exemption requested in attachment 1 ensures that failure of a reactor vessel will not occur. Therefore, a failure of the reactor vessel can still be considered incredible.

The proposed heatup and cooldown limit curves have been constructed by combining the most conservative pressure-temperature limits derived by using material properties of the intermediate forging, closure head flange, and beltline circumferential weld to form a single set of composite curves. Use of the proposed curves, does not modify the reactor coolant system pressure boundary, nor make any physical changes to the LTOP setpoint or design. Proposed Figures TS 3.1-1 and TS 3.1-2 were prepared in accordance with regulatory and code requirements and were derived using conservative material property basis and neutron exposure projections thru 33 EFPY. Therefore, the proposed heatup and cooldown curves and LTOP limits will continue to protect the reactor vessel from failure.

The LTOP system enabling temperature and the proposed Appendix G pressure temperature limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. The changes do not cause the initiation of any accident nor create any new credible limiting failure for safety-related systems and components. The changes do not result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than previously evaluated. The changes do not have any adverse effect on the ability of the safety-related systems to perform their intended safety functions.

The proposed changes do not make physical changes to the plant or create new failure modes. Thus, the PA does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed Appendix G pressure temperature limitations and corresponding LTOP enabling temperature were prepared using methods derived from the ASME Boiler and Pressure Vessel Code, including ASME Code Cases N-514 , N-588, and N-629.

Inherent conservatism in the P/T limits resulting from these documents is described in the Safety Evaluation.

Alternative methodologies to the safety margins required by Appendix G to 10CFR Part 50 have been developed by the ASME Working Group on Operating Plant Criteria. Three of these methodologies are contained in ASME Code Cases N-514, N-588, and N-629.

Code Case N-514 provides criteria to determine pressure limits during LTOP events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of the relief valve used for LTOP. Specifically, the ASME Code Case N-514 allows determination of the setpoint for LTOP events such that the maximum pressure in the vessel would not exceed 110% of the P/T limits of the existing ASME Appendix G; and redefines the enabling temperature at a coolant temperature less than 200°F or a reactor vessel metal

temperature less than RT_{NDT} + 50°F, whichever is greater. Code Case N-514, "Low Temperature Overpressure Protection," has been approved by the ASME Code Committee but not yet approved for use in Regulatory Guides 1.147. 1.85, or 1.84. The content of this Code Case has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. It is expected that the next revision of 10 CFR 50.55a will endorse the 1993 Addenda and Appendix G of Section XI. Code Case N-514 is not in conflict with 10CFR 50.61 and therefore has been used to establish the LTOP system enabling temperature; the provision for exceeding 110% of the Appendix G limits has not been incorporated in PA 160. The NRC previously approved use of Code Case N-514 for determination of the LTOP enabling temperature in reference 6.

Code Case N-588 provides benefits in terms of calculating pressure-temperature limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The NRC previously approved use of Code Case N-588 for use at KNPP in references 4 and 5.

In support of this PA, WPSC used fracture toughness results representing the beltline weld metal that were irradiated to EOL and in excess of EOLE fluence. The fracture toughness results were analyzed as described under Case #6 in WCAP-15075 and ASME Code Case N-629 for determining the EOL and EOLE indexing reference temperature values. Attachment 1 to this letter provides information to support NRC approval to use the weld metal fracture toughness results along with the methodology presented in WCAP-15075 for the KNPP PTS evaluation. The KNPP application of the methodology presented in WCAP-15075, identified as Case #6, incorporates the following additional margins beyond that recommended in ASTM E1921-97:

a) A delta value of 17°F is added to T_0 to ensure that the margin in the KNPP application is at least as conservative as the margin associated with the most limiting HSST-02 plate material.

b) An additional margin of 18°F has been added to the above 17°F to be consistent with the ASME Code Case N-629, and align the KNPP lead plant application with current consensus of the technical community regarding the best use of fracture toughness based indexing reference temperature data.

c) A 2σ value of 16°F and 24°F is added to account for RT_{T_0} measurement uncertainty for EOL and EOLE, respectively.

d) A value of (+)35°F and (-)32°F accounts for heat uncertainty between the KNPP and Maine Yankee surveillance capsule specimens for EOL and EOLE, respectively.

Fracture toughness testing of irradiated 1P3571 weld metal, performed in accordance with ASTM E1921-97 and application of ASME Code Case N-629 along with the methods in

WCAP-15075, indicate that the end of life indexing reference temperature is 234°F. This fracture toughness generated EOL indexing reference temperature value includes a margin of 34°F (18°F + 16°F). The fracture toughness generated indexing reference temperature value (234 °F) is lower than the ART value (277°F) predicted by the Charpy V-notch and Drop Weight methodology. Both methodologies predict end of life indexing reference temperature values that are below the pressurized thermal shock screening criteria (300°F).

Use of the methodology set forth in the ASME Boiler and Pressure Vessel Code, NRC Regulatory Standard Review Plan 5.3.2., WCAP-15075, 10 CFR 50.61, and 10 CFR 50 Appendices G and H ensures that proper limits and safety factors are maintained. Thus, the PA does not involve a significant reduction in the margin of safety.

The revised heatup and cooldown limit curves and corresponding LTOP system enabling temperature were prepared using fracture toughness, drop weight and Charpy V-notch data for the beltline weld material; drop weight and Charpy V-notch data for the closure head flange and intermediated forging material; along with practices described herein and methods derived from the ASME Boiler and Pressure Vessel Code and 10 CFR 50.61. The safety factors and margins used in the development of the limit curves and LTOP system enabling temperature meet the criteria set forth by these documents. Application of low leakage core designs decreases the rate of shift in transition temperature from ductile to nonductile behavior. The revised limit curves and corresponding LTOP enabling temperature provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, and low temperature overpressure protection (corresponding to isothermal events during low temperature operations (i.e., ≤ 200°F)). With the preparation of the revised limit curves in accordance with the latest criteria and guidance, this PA ensures that proper limits and safety factors are maintained.

Thus, the PA does not involve a significant reduction in a margin of safety. Therefore, the proposed amendment does not represent a significant decrease in the margin of safety. As shown in attachment 1, a loss of reactor vessel integrity is still incredible. Furthermore, the LTOP setpoint and enabling temperature will continue to protect the reactor coolant system during low temperature operation.

ENVIRONMENT CONSIDERATIONS

This PA involves a change to a requirement with respect to the use of a facility component located within the restricted area, as defined in 10 CFR Part 20. WPSC has determined that the PA involves no significant hazards considerations and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this PA meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this PA.

ATTACHMENT 3

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 7, 1999

Proposed Amendment 160

Affected TS Pages:

Table of Contents - List of Figures (TS vi)
TS 3.1-6
TS B3.1-4
TS B3.1-6 thru TS B3.1-10
TS B3.1-12 thru TS B3.1-14
Figure TS 3.1-1
Figure TS 3.1-2

LIST OF FIGURES

b. Heatup and Cooldown Limit Curves for Normal Operation

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 TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33 effective full-power years.

   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.

   B. Figures TS 3.1-1 and TS 3.1-2 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.

   C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

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

Neatup and Cooldown Limit Curves for Normal Operation (TS 3.1.b)

Fracture Toughness Properties - (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,[5] and the calculation methods of Footnote.[6] The postirradiation fracture toughness properties of the reactor vessel belt line materials were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program in accordance with ASTM E185 and ASTM E1921.

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, and are discussed in detail in Footnote.[7]

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

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[5] Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

[6] Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

[7] WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

With NRC approval to use ASME Code Case N-588 and ASME Code Case N-629, the reactor vessel intermediate forging and head flange become the controlling materials for development of the heatup and cooldown limit curves. Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above, ASME Boiler and Pressure Vessel Code Case N-588, and ASME Boiler and Pressure Vessel Code Case N-629. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan,[8] Footnote,[9] and WCAP-15075.[10]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,[11] weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

Fracture toughness testing of reconstituted Charpy V-notch specimens, obtained from the KNPP Radiation Surveillance Program, has been performed to determine the adjusted reference temperature of the core region weld metal corresponding to end of life fluence.

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[8] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[9] 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

[10] WCAP-15075, "Master Curve Strategies for RPV Assessment," R.G. Lott and M.T. Kirk, September 1998.

[11] WCAP-14279, Revision 1, "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," C. Kim, et al., September 1998.

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,[12] WCAP 9878,[13] WCAP-12020,[14] WCAP-14279,[15] and WCAP-14279, Revision 1[11] respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.I-1 and TS 3.1-2 and represent an operational time period of 33 effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33 effective full-power years of fluence (through the end of operating cycle 33). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Pressurizer Limits - (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

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[12] WCAP 8908, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, and K.V. Scott, January 1977.

[13] WCAP 9878, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, et al., March 1981.

[14] WCAP-12020, "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, et al., November 1988.

[15] WCAP-14279, "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," E. Terek, et al., March 1995.

Low Temperature Overpressure Protection - (TS 3.1.b.4)

The low temperature overpressure protection system must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33 effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F and the head is on the reactor vessel. The LTOP system is considered operable when all 4 valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within 5 days. If the isolated flowpath cannot be restored within 5 days, the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional 8 hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, the system can still be considered operable if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), the vent path is considered secured in the open position.

Maximum Coolant Activity (TS 3.1.c)

The limit on gross specific activity is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[116] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.[116]

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose, rem = 1/2 \quad [\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10}) \quad (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\bar{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9 x 10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

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[116] USAR Section 14.2.4

Reactor coolant specific activity is further limited to ≤ 0.20 μCi/gram DOSE EQUIVALENT I-131 to ensure that off-site thyroid dose does not exceed 10 CFR 100 guidelines and that the control room thyroid dose does not exceed GDC-19. To ensure the allowable doses are not exceeded, an evaluation was performed to determine the maximum allowable primary to secondary leak rate which could exist during a steam line break event. This analysis is described in the Basis for TS 3.4.d on secondary activity limits.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 0.20 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown in Figure TS 3.1-3, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing average coolant to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

Leakage of Reactor Coolant (TS 3.1.d)[47]

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

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[47] USAR Sections 6.5, 11.2.3, 14.2.4

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[18]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.[19] Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

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[18] USAR Section 4.2

[19] USAR Section 9.2

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at ≤ 60% RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.[20][21]

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[20]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that the reactor is not to be made critical when the moderator coefficient is > 5.0 pcm/°F has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than 5.0 pcm/°F provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[22] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

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[20] USAR Table 3.2-1

[21] USAR Figure 3.2-8

[22] USAR Figure 3.2-9

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality ≤ 5.0 pcm/°F will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports operating up to 60% power with a moderator temperature coefficient ≤ 5.0 pcm/°F. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than -8.0 pcm/°F for at least 95% of a cycle's time at HFP to ensure the limitations associated with and Anticipated Transient Without Scram (ATWS) event are not exceeded. NRC approved methods[23][24] will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than -8.0 pcm/°F, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

In the event that the limits of TS 3.1.f.3 are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit in TS 3.1.f.3. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

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[23] "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

[24] "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

FIGURE TS 3.1-1

KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES

APPLICABLE FOR PERIODS UP TO 33 EFFECTIVE FULL-POWER YEARS

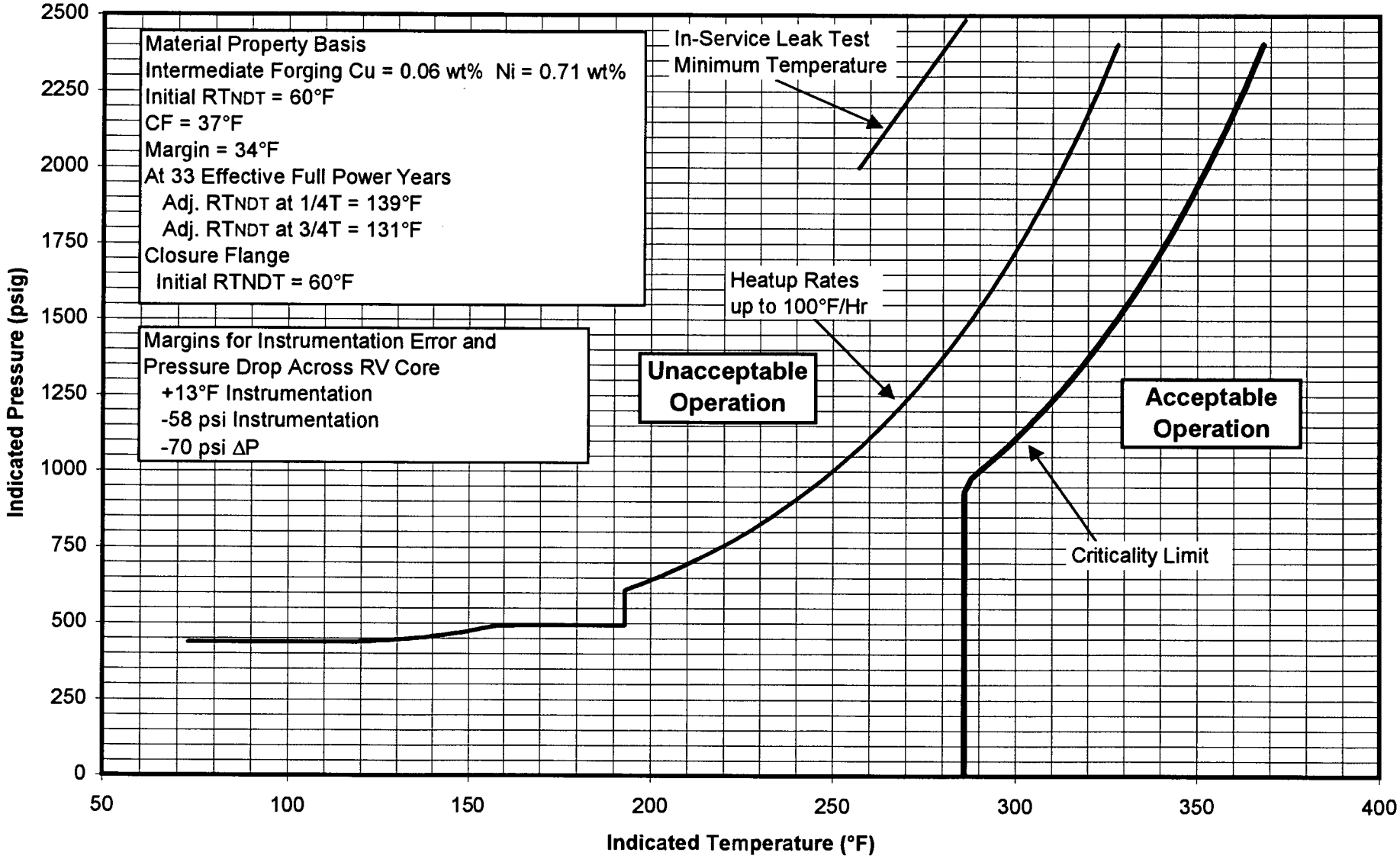

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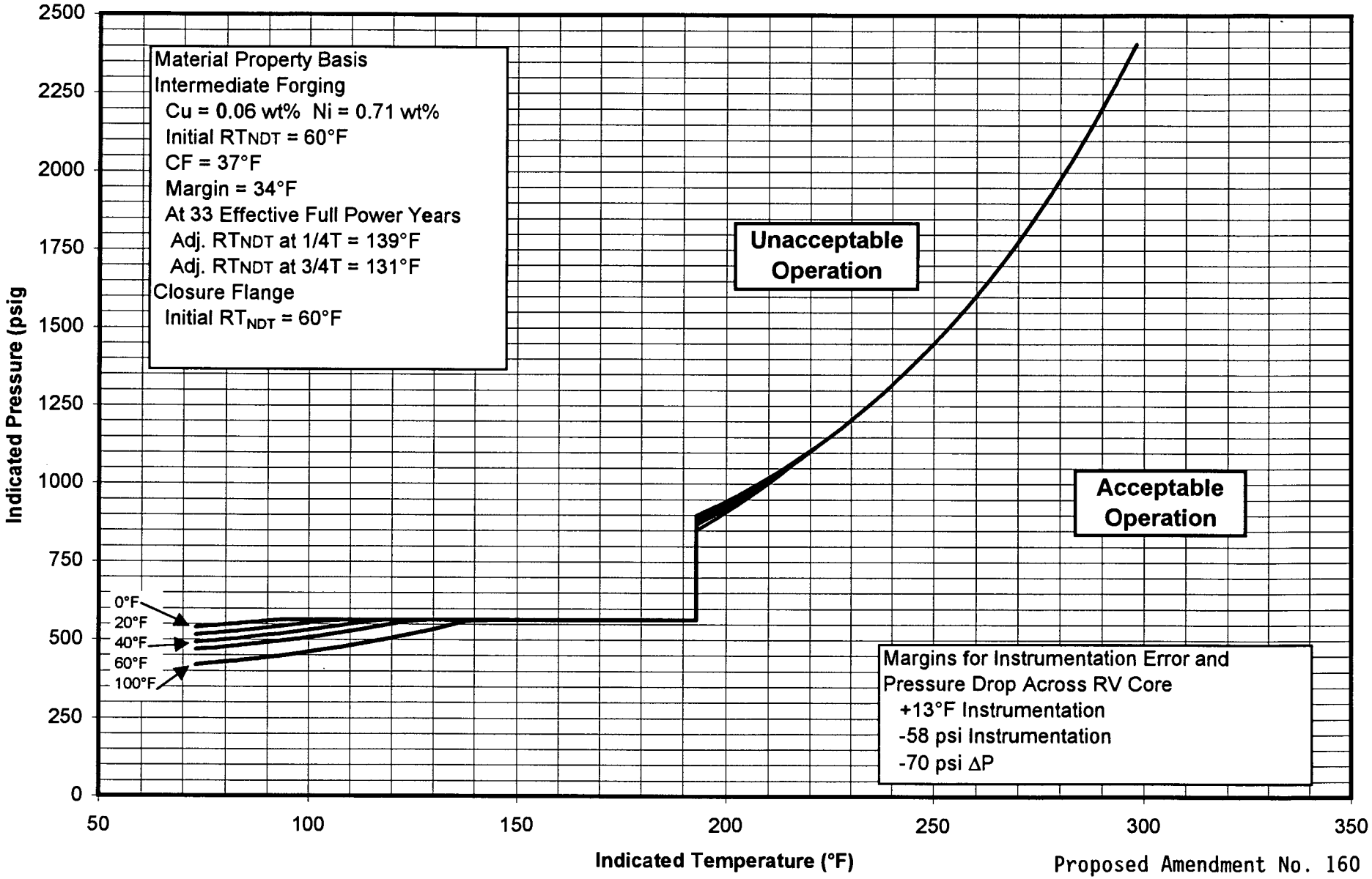
FIGURE TS 3.1-2

KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES

APPLICABLE FOR PERIODS UP TO 33 EFFECTIVE FULL-POWER YEARS



Proposed Amendment No. 160
06/07/99

ATTACHMENT 4

Letter from M.L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

June 7, 1999

Proposed Amendment 160

Strikeout TS Pages:

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TS B3.1-12 thru TS B3.1-14
Figure TS 3.1-1
Figure TS 3.1-2

LIST OF FIGURES

~~Note:~~

~~[111] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

b. Heatup and Cooldown Limit Curves for Normal Operation

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 TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33[~~111~~] effective full-power years.

   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.

   B. Figures TS 3.1-1 and TS 3.1-2 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.

   C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

2. The secondary side of the steam generator must not be pressurized > 200 psig if the temperature of the steam generator is < 70°F.

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

~~Note:~~

~~[111] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

Heatup and Cooldown Limit Curves for Hormal Operation (TS 3.1.b)

Fracture Toughness Properties - (TS 3.1.b.1)

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the ASME Boiler and Pressure Vessel Code,[5] and the calculation methods of Footnote.[6] The postirradiation fracture toughness properties of the reactor vessel belt line materials were obtained directly from the Kewaunee Reactor Vessel Material Surveillance Program in accordance with ASTM E185 and ASTM E1921.

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, and are discussed in detail in Footnote.[7]

The method specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by the pressure gradient. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{Im} + K_{It} \leq K_{IR} \qquad (3.1b\text{-}1)$$

where

K_{Im} is the stress intensity factor caused by membrane (pressure) stress

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

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[5] Section III and XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Protection Against Non-ductile Failure."

[6] Standard Method for Measuring Thermal Neutron Flux by Radioactivation Techniques, ASTM designation E262-86.

[7] WCAP-14278, Revision 1, "Kewaunee Heatup and Cooldown Limit Curves for Normal Operation," T. Laubham and C. Kim, September 1998.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady-state under certain conditions.

With NRC approval to use ASME Code Case N-588 and ASME Code Case N-629, the reactor vessel intermediate forging and head flange become the controlling materials for development of the heatup and cooldown limit curves. Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed above,~~and limited application to~~ ASME Boiler and Pressure Vessel Code Case N-588, ~~to the circumferential beltline weld~~and ASME Boiler and Pressure Vessel Code Case N-629. The derivation of the limit curves is consistent with the NRC Regulatory Standard Review Plan,[8] ~~and~~ Footnote~~.~~,[9] and WCAP-15075.[10]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. As presented in WCAP 14279, Revision 1,~~[10]~~[11] weld metal Charpy test specimens from Capsule S indicate that the core region weld metal exhibits the largest shift in RT_{NDT} (250°F).

Fracture toughness testing of reconstituted Charpy V-notch specimens, obtained from the KNPP Radiation Surveillance Program, has been performed to determine the adjusted reference temperature of the core region weld metal corresponding to end of life fluence.

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[8] "Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[9] 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

[10] WCAP-15075, "Master Curve Strategies for RPV Assessment," R.G. Lott and M.T. Kirk, September 1998.

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[11] WCAP-14279, Revision 1, "Evaluation of Capsule S from the Kewaunee and Capsule A35 from the Maine Yankee Nuclear Power Reactor Vessel Radiation Surveillance Programs," C. Kim, et al., September 1998.

The results of Irradiation Capsules V, R, P, and S analyses are presented in WCAP 8908,[~~11~~][12] WCAP 9878,[~~12~~][13] WCAP-12020,[~~13~~][14] WCAP-14279,[~~14~~][15] and WCAP-14279, Revision 1[~~10~~][11] respectively. Heatup and cooldown limit curves for normal operation of the reactor vessel are presented in Figures TS 3.1-1 and TS 3.1-2 and represent an operational time period of 33[~~1~~] effective full-power years.

The isothermal cooldown limit curve (Figure TS 3.1-2) is used for evaluation of low temperature overpressure protection (LTOP) events. This curve is applicable for 33[~~1~~] effective full-power years of fluence (through the end of operating cycle 33[~~1~~]). If a low temperature overpressure event occurred, the RCS pressure transient would be evaluated to the limits of this figure to verify the integrity of the reactor vessel. If these limits are not exceeded, vessel integrity is assured and a TS violation has not occurred.

Pressurizer Limits - (TS 3.1.b.3)

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with Code requirements. In-plant testing and calculations have shown that a pressurizer heatup rate of 100°F/hr cannot be achieved with the installed equipment.

~~Note:~~

~~[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

~~(11)~~

[12] WCAP 8908, "Analysis of Capsule V from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, S.L. Anderson, and K.V. Scott, January 1977.

~~(12)~~

[13] WCAP 9878, "Analysis of Capsule R from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, et al., March 1981.

~~(13)~~

[14] WCAP-12020, "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," S.E. Yanichko, et al., November 1988.

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Low Temperature Overpressure Protection - (TS 3.1.b.4)

The low temperature overpressure protection system must be OPERABLE during startup and shutdown conditions below the enable temperature (i.e., low temperature) as defined in Branch Technical Position RSB 5-2 as modified by ASME Boiler and Pressure Vessel Code Case N-514. Based on the Kewaunee Appendix G LTOP protection pressure-temperature limits calculated through 33[iii] effective full-power years, the LTOP System must be OPERABLE whenever one or more of the RCS cold leg temperatures are ≤ 200°F and the head is on the reactor vessel. The LTOP system is considered operable when all 4 valves on the RHR suction piping (valves RHR-1A, 1B, 2A, 2B) are open and valve RHR-33-1, the LTOP valve, is able to relieve RCS overpressure events without violating Figure TS 3.1-2.

The set pressure specified in TS 3.1.b.4 includes consideration for the opening pressure tolerance of ± 3% (± 15 psig) as defined in ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC: Class 2 Components for Safety Relief Valves. The analysis of pressure transient conditions has demonstrated acceptable relieving capability at the upper tolerance limit of 515 psig.

If one train of RHR suction piping to RHR 33-1 is isolated, the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within 5 days. If the isolated flowpath cannot be restored within 5 days, the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional 8 hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, the system can still be considered operable if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), the vent path is considered secured in the open position.

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The limit on gross specific activity is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

~~Note:~~

~~[iii] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[16] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.

The limiting off-site dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose, rem = 1/2 \quad [\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10}) \quad (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\bar{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9×10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

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[16] USAR Section 14.2.4

Reactor coolant specific activity is further limited to ≤ 0.20 μCi/gram DOSE EQUIVALENT I-131 to ensure that off-site thyroid dose does not exceed 10 CFR 100 guidelines and that the control room thyroid dose does not exceed GDC-19. To ensure the allowable doses are not exceeded, an evaluation was performed to determine the maximum allowable primary to secondary leak rate which could exist during a steam line break event. This analysis is described in the Basis for TS 3.4.d on secondary activity limits.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 0.20 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown in Figure TS 3.1-3, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing average coolant to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

<u>Leakage of Reactor Coolant (TS 3.I.d)</u>[~~116~~][17]

<u>TS (TS 3.1.d.1)</u>

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

~~116~~

[17] USAR Sections 6.5, 11.2.3, 14.2.4

C. Humidity detection provides a backup to A. and B. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.

D. A leakage detection system is provided which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by fancoils of the Containment Air Cooling System and thus provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. The fancoil units drain to the containment sump, and all leakage collected by the containment sump will be pumped to the waste holdup tank. Pump running time will be monitored in the control room to indicate the quantity of leakage accumulated.

If leakage is to another closed system, it will be detected by the area and process radiation monitors and/or inventory control.

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[~~17~~ (18)]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.[~~18~~ (19)] Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

~~(17)~~

(18) USAR Section 4.2

~~(18)~~

(19) USAR Section 9.2

Minimum Conditions for Criticality (TS 3.I.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at ≤ 60% RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.[~~19~~] [~~20~~] [20] [21]

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[~~19~~] [20]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that the reactor is not to be made critical when the moderator coefficient is > 5.0 pcm/°F has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than 5.0 pcm/°F provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[~~21~~] [22] and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction in moderator density.

[~~19~~]

[20] USAR Table 3.2-1

[~~20~~]

[21] USAR Figure 3.2-8

[~~21~~]

[22] USAR Figure 3.2-9

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality ≤ 5.0 pcm/°F will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports operating up to 60% power with a moderator temperature coefficient ≤ 5.0 pcm/°F. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than -8.0 pcm/°F for at least 95% of a cycle's time at HFP to ensure the limitations associated with and Anticipated Transient Without Scram (ATWS) event are not exceeded. NRC approved methods[22][23][24] will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than -8.0 pcm/°F, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

In the event that the limits of TS 3.1.f.3 are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit in TS 3.1.f.3. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

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[22]

[23] "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

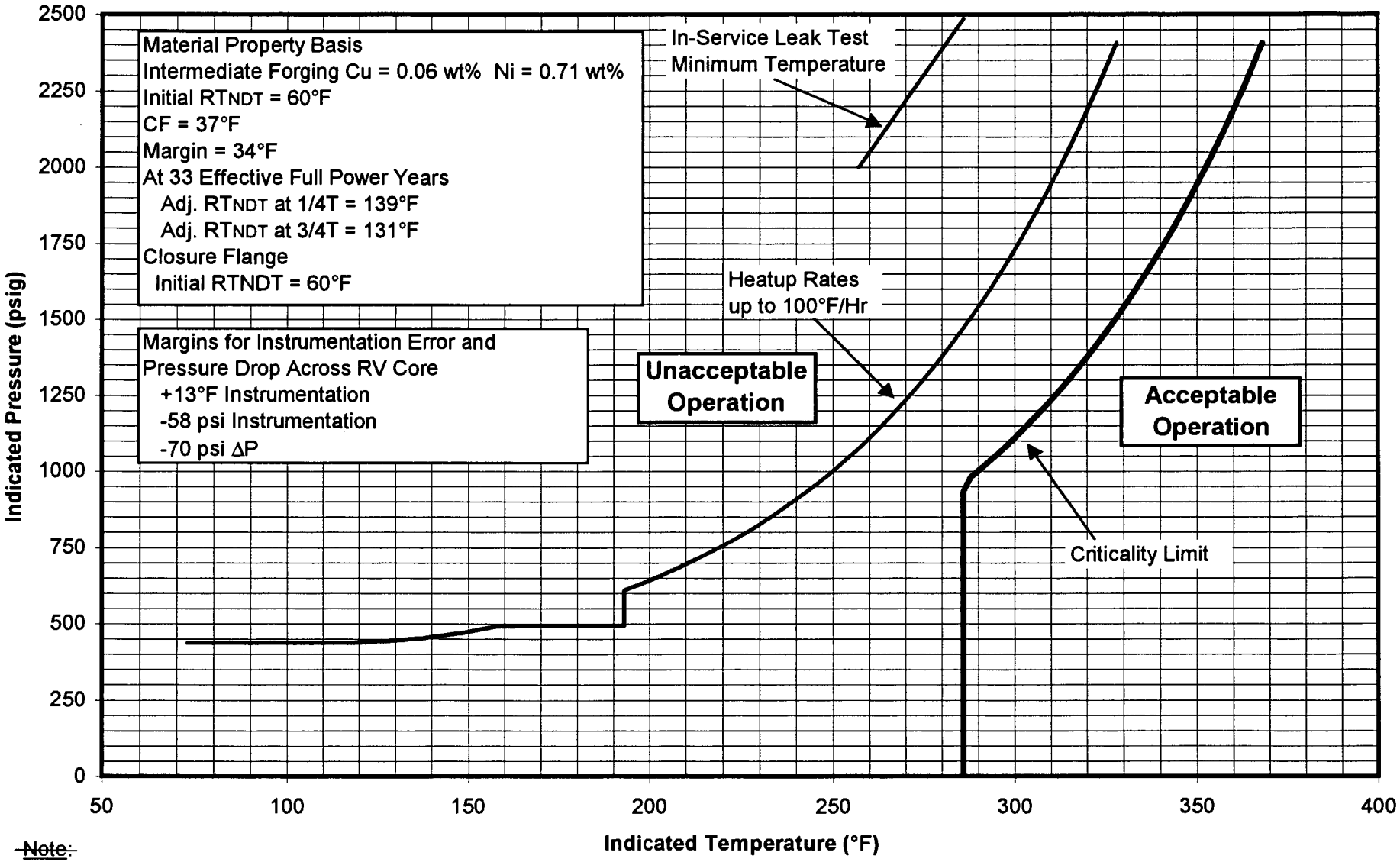
[24]

[24] "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

FIGURE ~~15~~ 3.1-1

KEWAUNEE UNIT NO. 1 HEATUP LIMITATION CURVES

APPLICABLE FOR PERIODS UP TO 33~~[1]~~ EFFECTIVE FULL-POWER YEARS



~~Note:~~

~~[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

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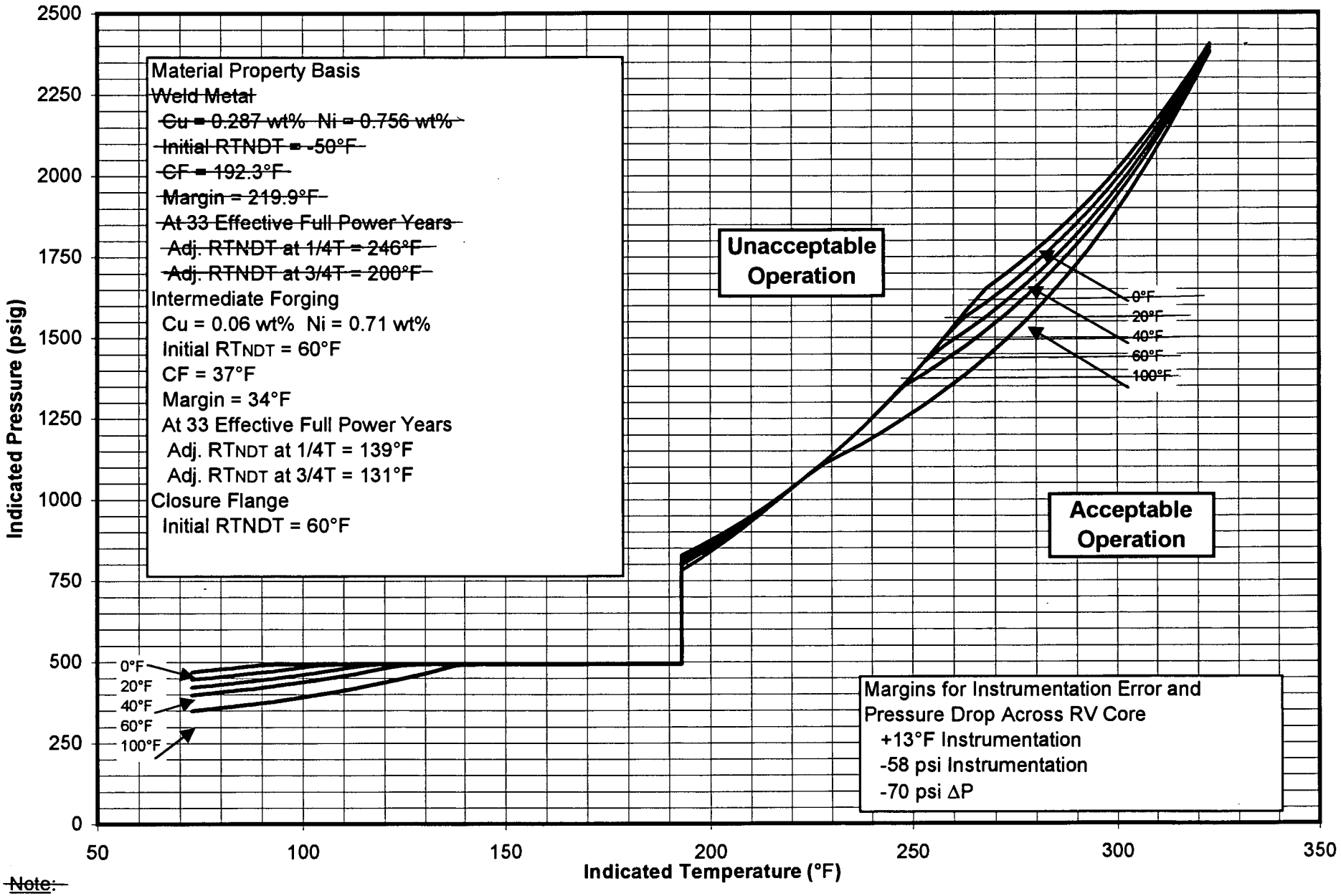
FIGURE 3.1-2

KEWAUNEE UNIT NO. 1 COOLDOWN LIMITATION CURVES

APPLICABLE FOR PERIODS UP TO 33[1] EFFECTIVE FULL-POWER YEARS



~~Note:~~

~~[1] Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by WPSC Letter NRC-99-017.~~

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