Duke Power
A Duke Energy Company

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March 20, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications (TS) Amendment
TS 3.4.3 - Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits
TS 3.4.6, RCS Loops - MODE 4
TS 3.4.7, RCS Loops - MODE 5, Loops Filled
TS 3.4.10, Pressurizer Safety Valves
TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)
TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System

Pursuant to 10 CFR 50.90, Duke Energy Corporation is requesting an amendment to the Catawba Nuclear Station Facility Operating License and Technical Specifications (TS). The proposed amendment revises TS 3.4.3 to update the heatup, cooldown, critically, and inservice test pressure and temperature (P/T) limits for the reactor coolant system (RCS) of each unit to a maximum of 34 Effective Full Power Years (EFPY). The current P/T limits are valid to 15 EFPY. Based on current projected operating cycles the existing P/T limits are expected to expire in March 2004 for Unit 1 and November 2005 for Unit 2. The changes to TS 3.4.3 are based on the analyses of latest reactor vessel capsule data and alternative methodology for determining P/T limits. The analyses of latest reactor vessel capsule data are documented in WCAP-15117, "Analysis of Capsule V and the Dosimeters from Capsules U and X from Duke Power Company Catawba Unit 1 Reactor Vessel Radiation Surveillance Program" and WCAP-15243, "Analysis of Capsule V and the Capsule Y Dosimeters from Duke Energy Catawba Unit 2 Reactor Vessel Radiation Surveillance Program". The alternative methodology for determining allowable P/T limits is described in American Society of Mechanical Engineers (ASME) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

A047
A001

The alternative methodology results in less restrictive P/T limits. This proposed amendment includes an exemption request pursuant to 10 CFR 50.12 from certain technical requirements of 10 CFR 50.60 and 10 CFR 50 Appendix G. The first technical exemption relies on ASME Code Case N-640 and is included as Attachment 5. A second technical exemption request relies on ASME Code Case N-641 to develop the LTOP enable temperature and is included as Attachment 6. The Nuclear Regulatory Commission (NRC) has approved exemptions and amendments associated with the use of these code cases in generating P/T limits at numerous nuclear power stations shown in Attachment 2.

The proposed amendment revises TS Bases to include a brief summary of the excore cavity dosimetry program to be installed at Catawba. The proposed amendment revises TS 3.4.6, RCS Loops - MODE 4, TS 3.4.7, RCS Loops - MODE 5, Loops Filled, TS 3.4.10, Pressurizer Safety Valves, TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs), to reflect a revision in the LTOP system enable temperature from 285 °F to 210 °F. The revised LTOP enable temperature being proposed was developed using methodology provided in ASME Code Case N-641. Use of ASME Code Case N-641 methodology in the determination of the LTOP enable temperature is more technically correct than the generic value included in earlier versions of ASME XI and eliminates inconsistencies in the margin of safety between reactor vessel geometries.

The proposed amendment revises TS 3.4.12, LTOP System, to reflect the revised P/T limits and the revised LTOP enable temperature, to allow credit for the residual heat removal system suction relief valves as pressure relieving devices for LTOP system, and to allow a maximum of two pumps capable of injecting into the reactor coolant system. Duke has performed calculations that demonstrate that the revised TS 3.4.12 requirements provide adequate RCS overpressure protection by having a minimum input capability and adequate pressure relief capacity.

The TS Bases have been revised appropriately to reflect the TS changes described above. The contents of this amendment request package are as follows:

- Attachment 1 provides a marked copy of the affected TS pages for Catawba, showing the proposed changes.

- Attachment 2 provides a description of the proposed changes and technical justification.

- Pursuant to 10 CFR 50.92, Attachment 3 documents the determination that the amendment contains No Significant Hazards Considerations.

- Pursuant to 10 CFR 51.22(c)(9), Attachment 4 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

- Attachment 5 provides the exemption request for ASME Code Case N-640.

- Attachment 6 provides the exemption request for ASME Code Case N-641.

- Attachment 7 contains Westinghouse Report WCAP-15203, Revision 1, for the proposed Catawba Unit 1 heatup and cooldown limit curves.

- Attachment 8 contains Westinghouse Report WCAP-15285, for the proposed Catawba Unit 2 heatup and cooldown limit curves.

Implementation of this amendment to the Catawba Facility Operating License and TS will impact the Catawba Updated Final Safety Analysis Report (UFSAR). Changes to the affected UFSAR will be made in accordance with 10 CFR 50.71(e). Completing these UFSAR changes is the only regulatory commitment associated with this amendment.

Duke is requesting a 90-day implementation period in conjunction with this amendment. Duke is requesting the 90 days due to the nature of the TS being revised and the associated procedure changes necessary for implementation. The exception to this is the excore cavity dosimetry program which will be implemented in the next refueling outage after TS approval by the NRC.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the Catawba Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

The material supplied in support of this amendment is detailed and lengthy. It may be appropriate to schedule a meeting between Duke and the NRC staff to outline this material early in the review process. Duke will consult with the NRC Project Manager in this regard.

Duke is requesting NRC review and approval of this proposed amendment by September 2003, so that it may be implemented in conjunction with the Catawba Unit 1 End-of-Cycle 14 Refueling Outage.

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to R. D. Hart at (803) 831-3622.

Very truly yours,

Gary R. Peterson

Gary R. Peterson

RDH/s

Attachments

Gary R. Peterson affirms that he the person who subscribed his name to the foregoing statement, and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Gary R. Peterson, Site Vice President

Subscribed and sworn to me: 3-20-03
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

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ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATIONS PAGES FOR CATAWBA

<u>MATERIAL PROPERTY BASIS</u>

LIMITING MATERIAL: Intermediate LOWER SHELL FORGING ~~04~~ 05 & 04
LIMITING ART AT ~~15~~ 34 EFPY: 1/4-T, ~~43~~°F 42
3/4-T, ~~26~~°F 31

Insert New Chart A

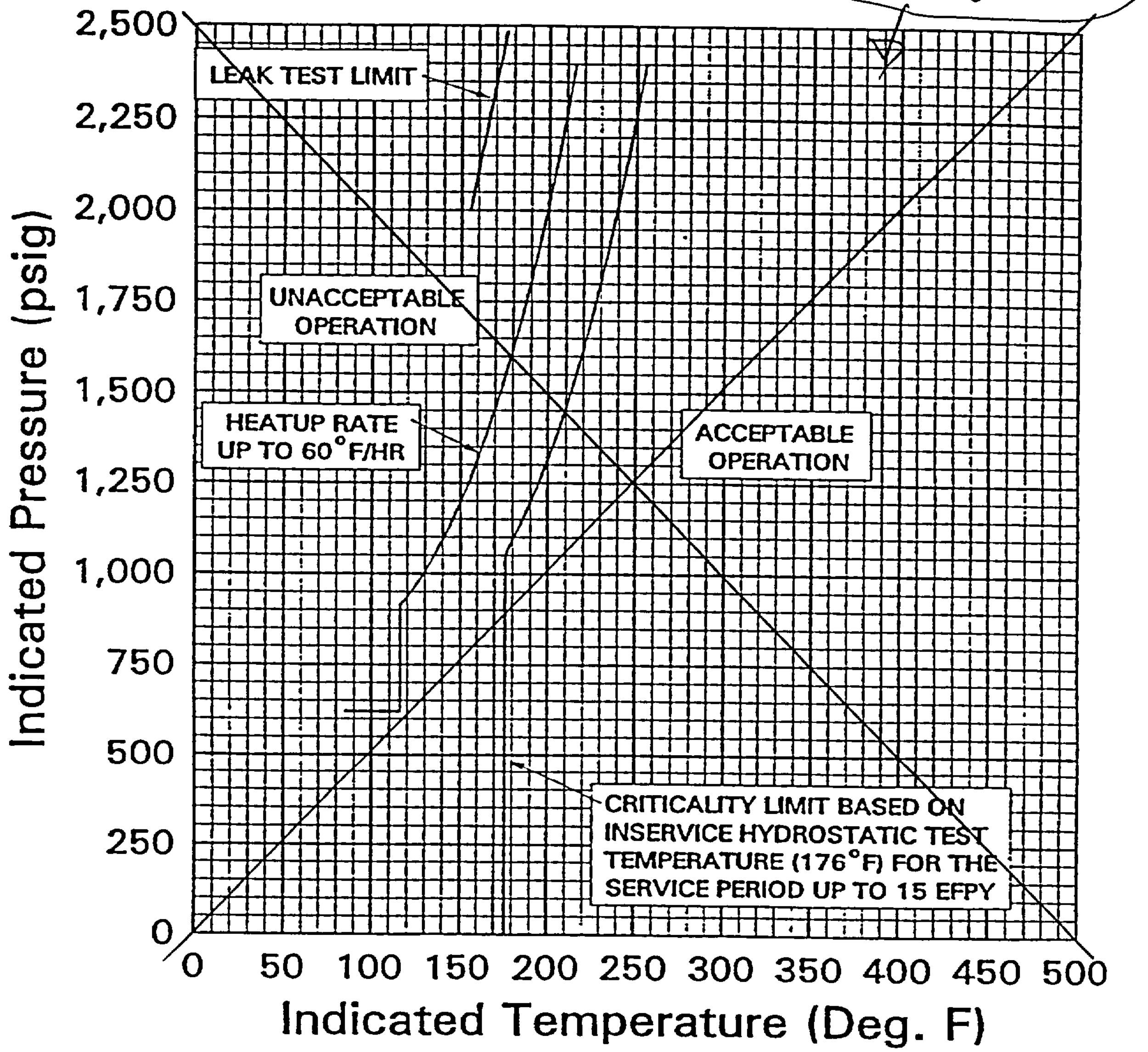

Figure 3.4.3-1
(UNIT 1 ONLY)
RCS Heatup Limitations

Chart A

MATERIALS PROPERTY BASIS
Limiting Material: Lower Shell Forging 04
and Intermediate Shell Forging 05
Limiting ART at 34 EFPY: 1/4-T, 42°F
3/4-T, 31°F

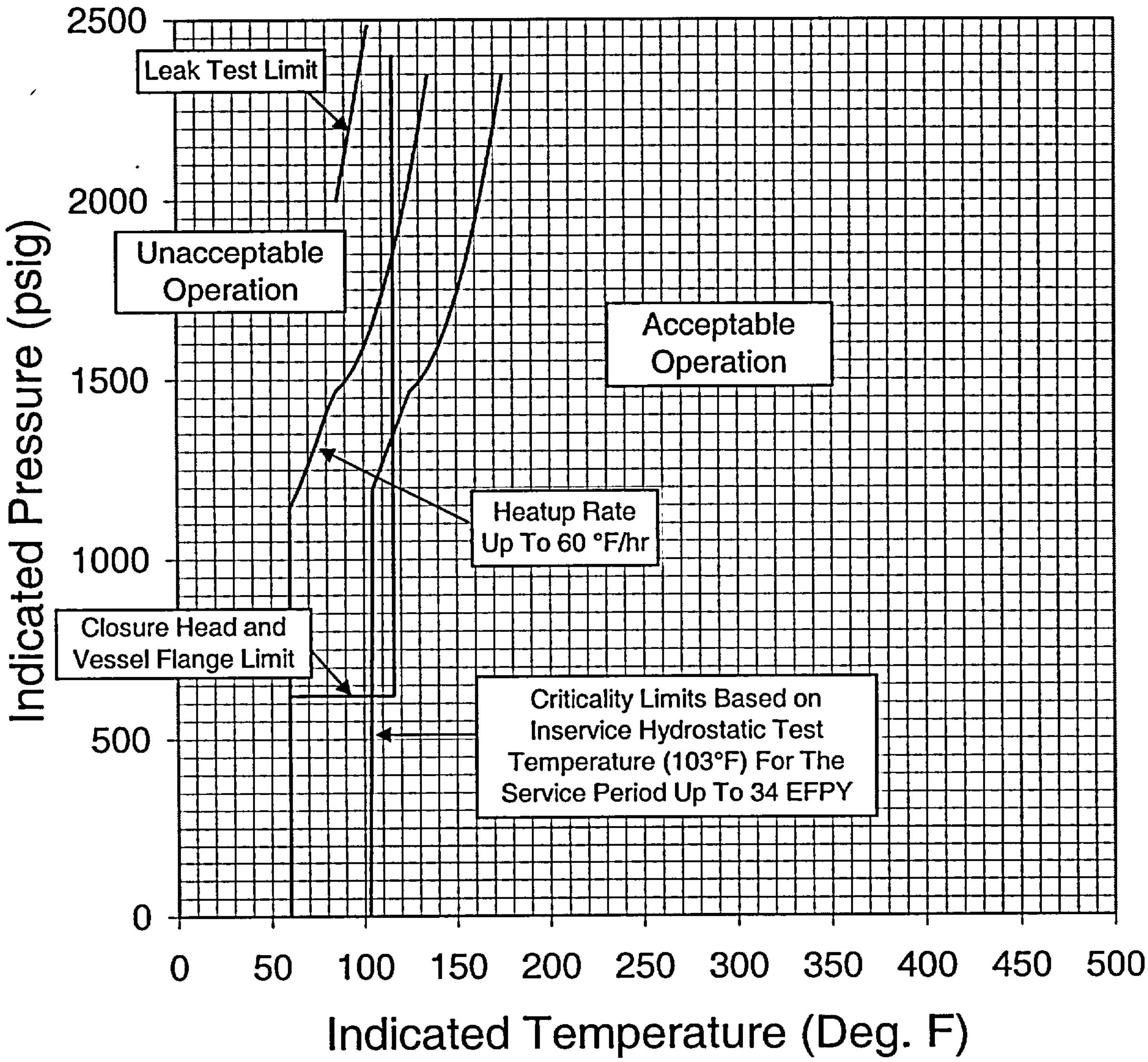

Figure 3.4.3-1
(UNIT 1 ONLY)
RCS Heatup Limitations

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: INTERMEDIATE SHELL, B8605-2
LIMITING ART AT ~~15~~ 34 EFPY: 1/4-t, ~~112.6~~ 121 °F
3/4-t, ~~96.8~~ 106 °F

Insert New Chart B

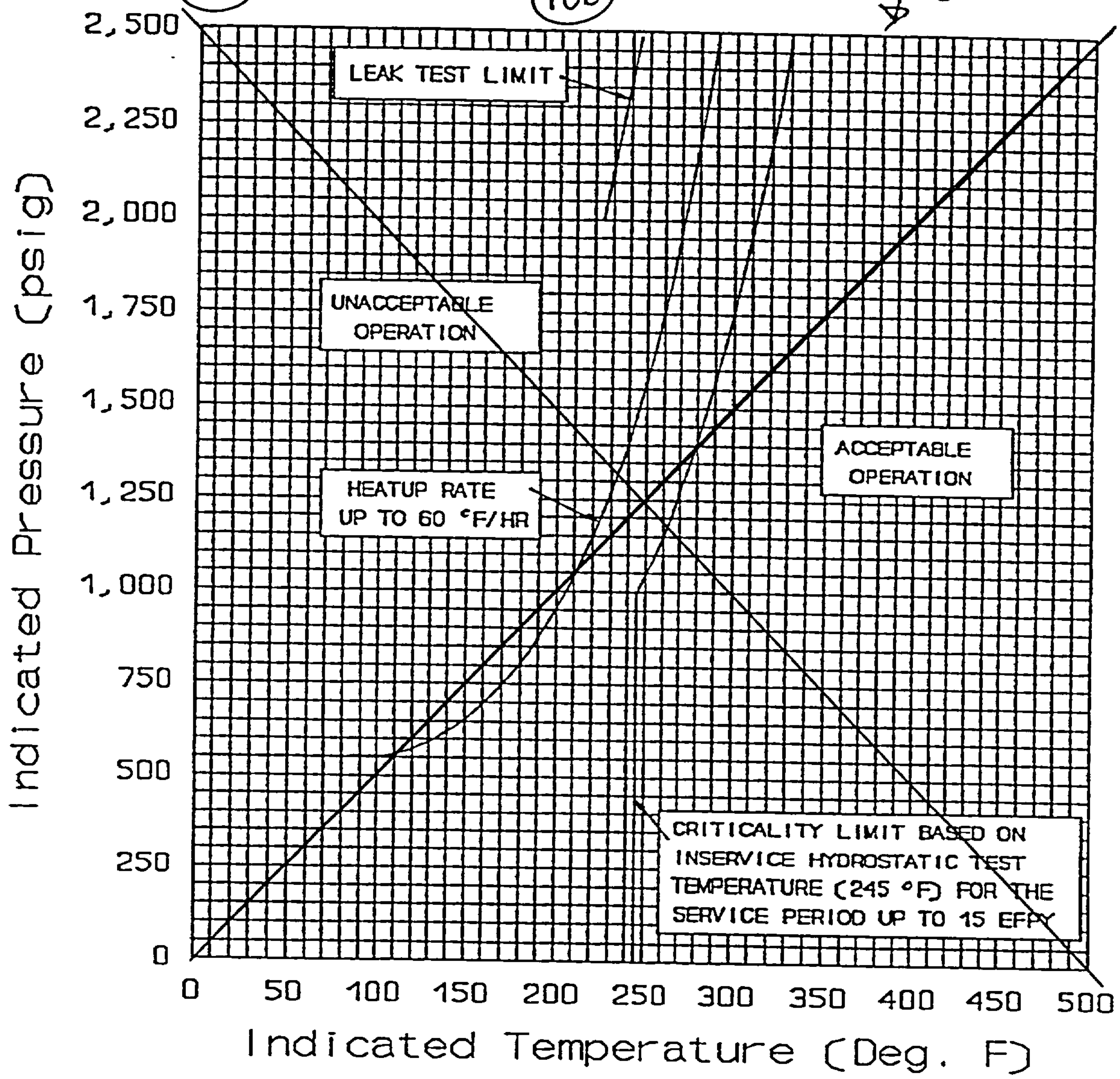

Figure 3.4.3-1
(UNIT 2 ONLY)
RCS Heatup Limitations

Chart 3

MATERIALS PROPERTY BASIS
Limiting Material: Intermediate Shell, B8605-2
Limiting ART at 34 EFPY: 1/4-T, 121°F
3/4-T, 106°F

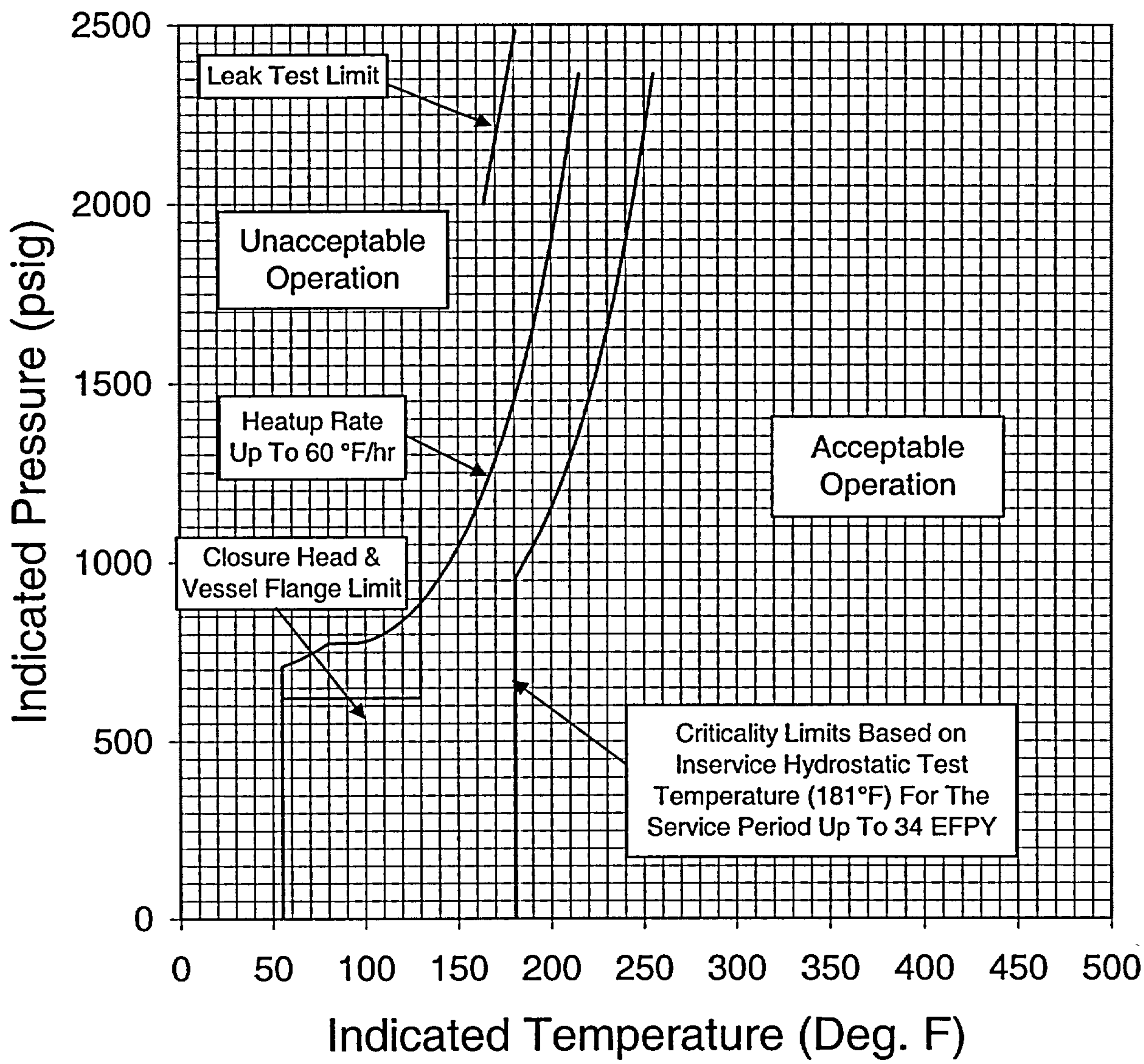

Figure 3.4.3-1
(UNIT 2 ONLY)
RCS Heatup Limitations

Intermediate &

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING ~~04~~ 05 & 04
LIMITING ART AT ~~15~~ 34 EFPY: 1/4-T, ~~43~~°F 42
3/4-T, ~~26~~°F 31

Insert New Chart C

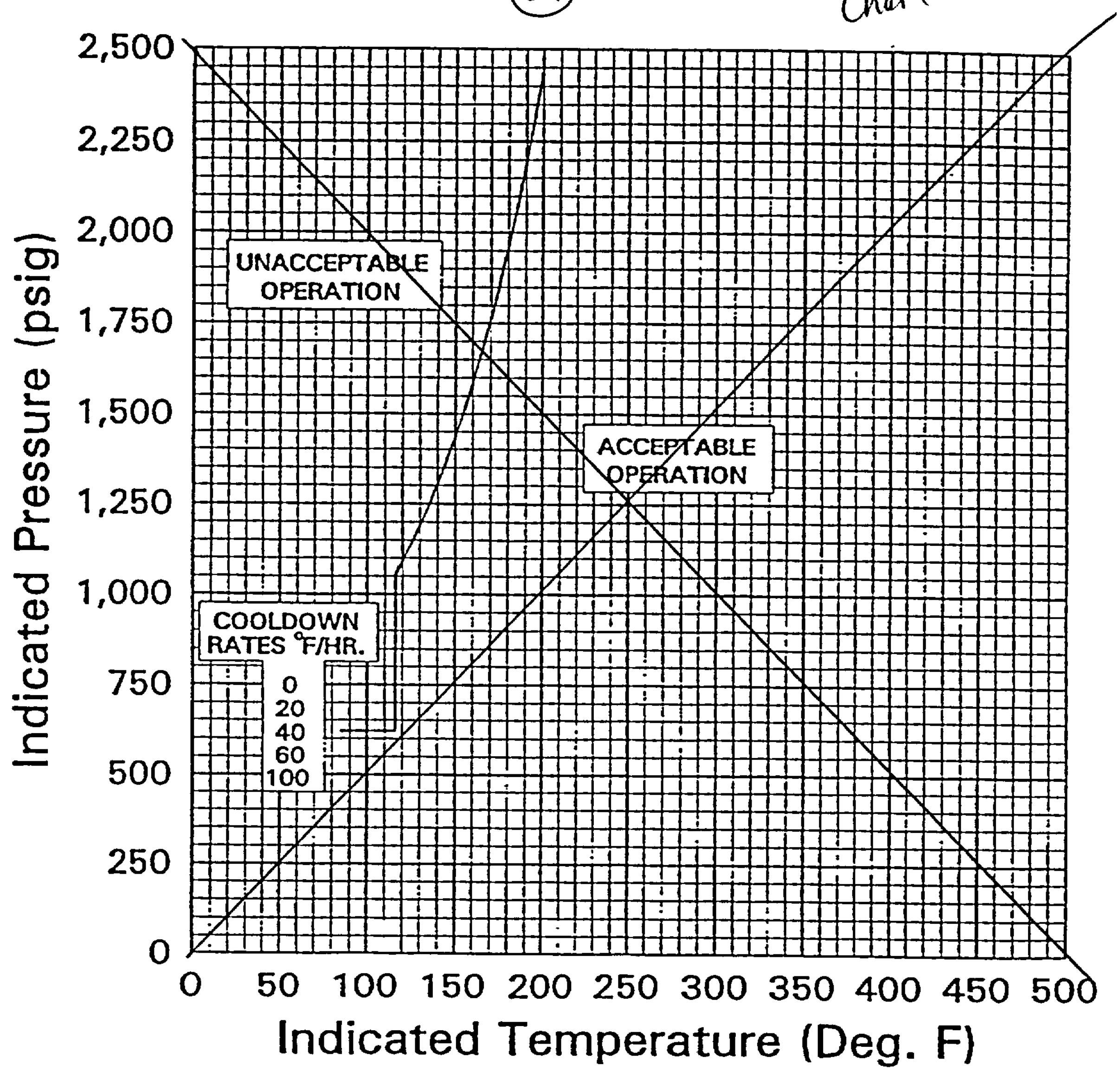

Figure 3.4.3-2
(UNIT 1 ONLY)
RCS Cooldown Limitations

Chart C

MATERIALS PROPERTY BASIS
Limiting Material: Lower Shell Forging 04
and Intermediate Shell Forging 05
Limiting ART at 34 EFPY: 1/4-T, 42°F
3/4-T, 31°F

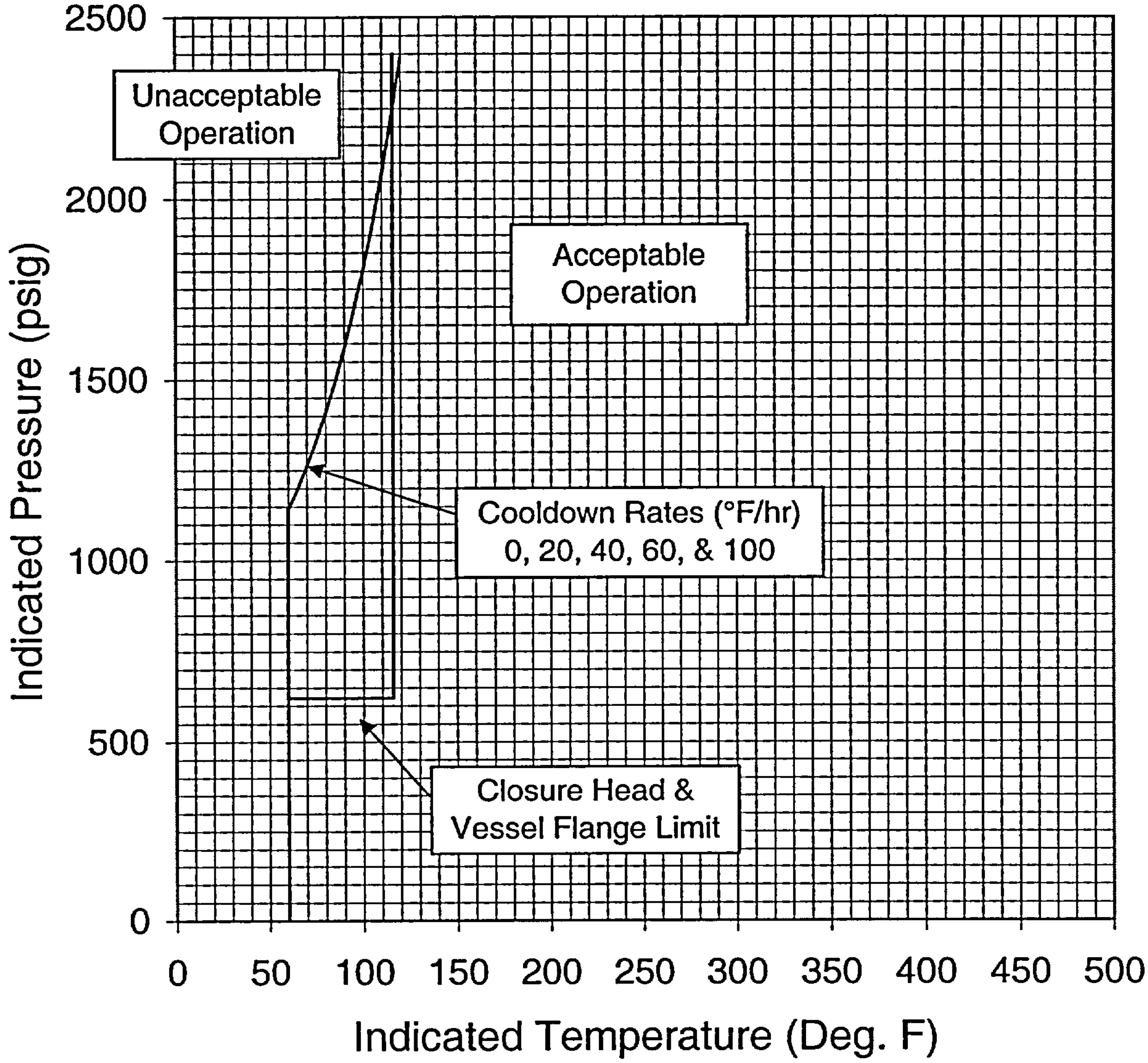

Figure 3.4.3-2
(UNIT 1 ONLY)
RCS Cooldown Limitations

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: INTERMEDIATE SHELL, B8605-2
LIMITING ART AT ~~15~~ 34 EFPY: 1/4-t, ~~112.6~~ 121 °F
3/4-t, ~~96.0~~ 106 °F

Insert New Chart D

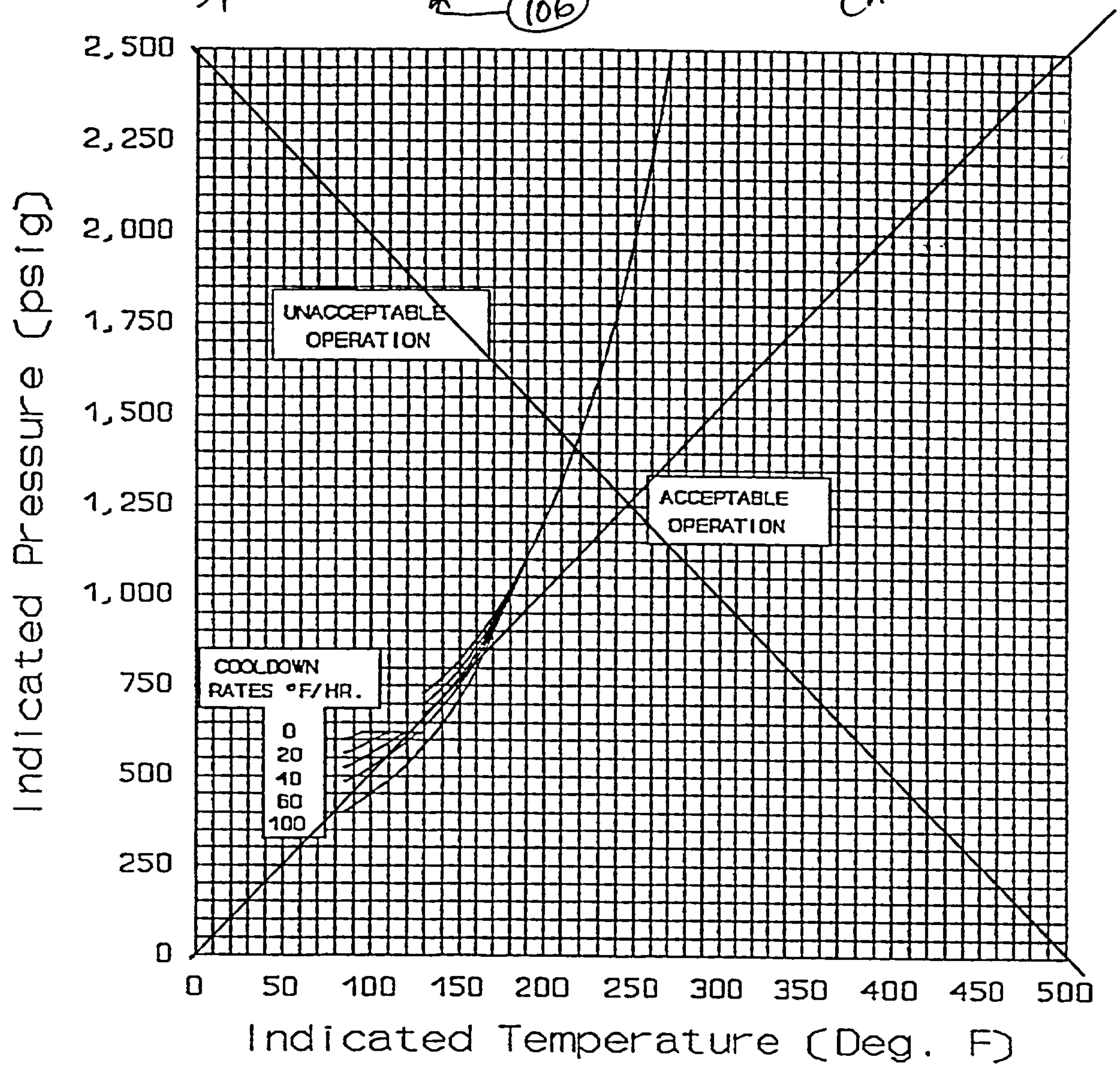

Figure 3.4.3-2
(UNIT 2 ONLY)
RCS Cooldown Limitations

Chart D

MATERIALS PROPERTY BASIS
Limiting Material: Intermediate Shell, B8605-2
Limiting ART at 34 EFPY: 1/4-T, 121°F
3/4-T, 106°F

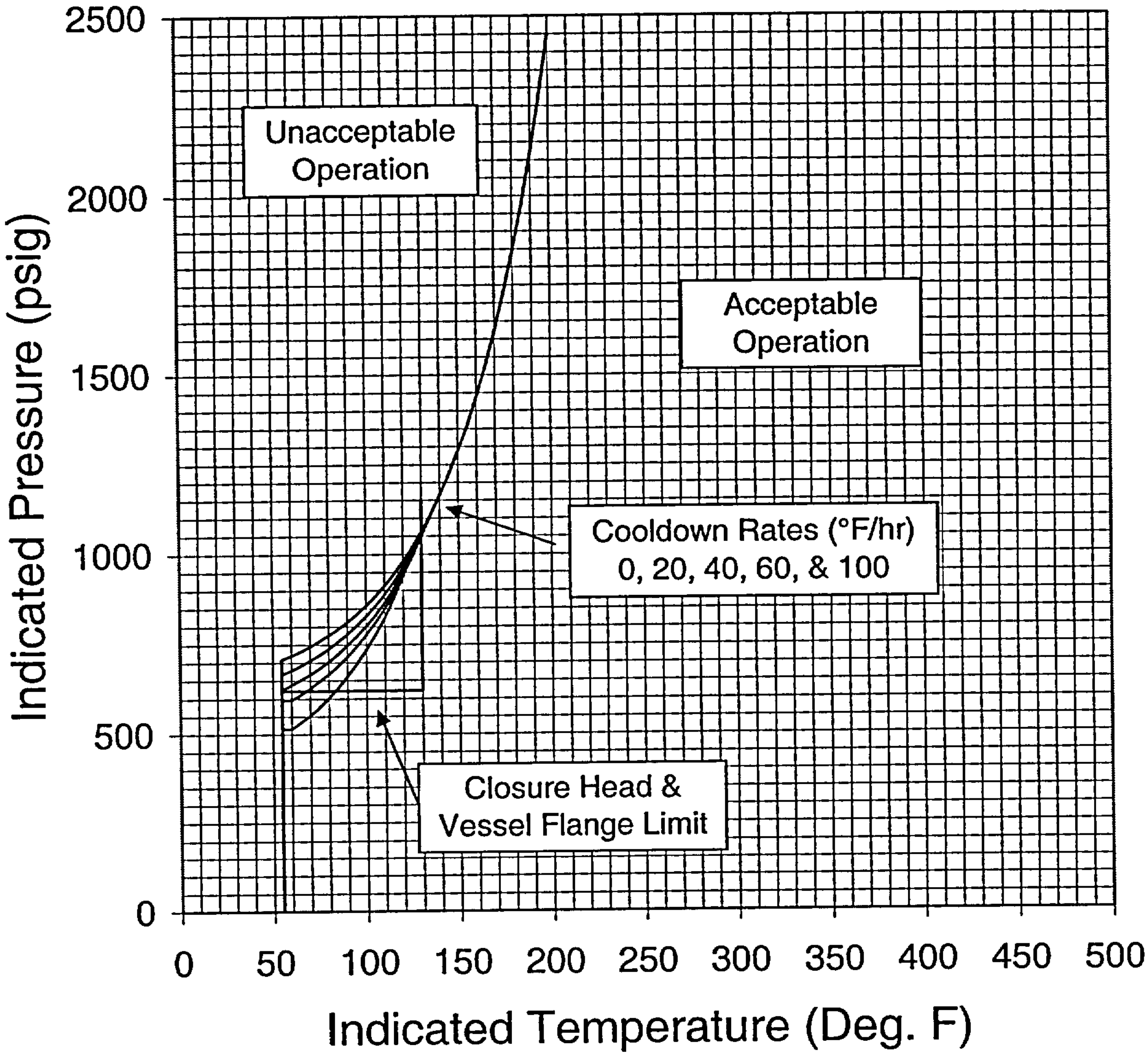

Figure 3.4.3-2
(UNIT 2 ONLY)
RCS Cooldown Limitations

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops — MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

--------------------------------NOTES--------------------------------

1. All reactor coolant pumps (RCPs) and RHR pumps may be de-energized for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause reduction of the RCS boron concentration; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature ≤ ~~285~~ 210°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop OPERABLE. AND Two RHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops — MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be $\geq$ 12% narrow range.

-------------------------------------NOTES-------------------------------------

1. The RHR pump of the loop in operation may be de-energized for $\leq$ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures $\leq$ ~~285~~ 210°F unless the secondary side water temperature of each SG is $\leq$ 50°F above each of the RCS cold leg temperatures.

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2435 psig and $\leq$ 2559 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > ~~285~~ 210°F.

------------------------------NOTE------------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperatures $\leq$ ~~285~~°F. 210	6 hours 12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

or charging and safety injection pumps

two pumps

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of ~~one~~ charging pump ~~or one~~ safety injection pump capable of injecting into the RCS, the accumulators isolated, reactor coolant pump operation limited as specified in Table 3.4.12-1 and either a, ~~or~~ b, or c below:

a. Two power operated relief valves (PORVs) with nominal lift setting = 400 psig (as left calibrated), allowable value ≤ 425 psig (as found), with RCS cold leg temperature ≥ ~~65~~ 70°F; or

~~b. The RCS depressurized and an RCS vent of ≥ 4.5 square inches.~~

b. Two residual heat removal (RHR) suction relief valves with lift settings ≥ 417 psig and ≤ 509 psig with an indicated RCS cold leg temperature ≥ 70 °F; or

c. A combination of any one PORV and one RHR suction relief valve, each with lift settings as described above.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~285~~ 210°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

---NOTE---
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in Specification 3.4.3.

ACTIONS

---NOTE---

LCO 3.0.4 is not applicable.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	More than two pumps (charging, safety injection, or charging and safety injection) capable of injecting into the RCS. ~~Two or more charging pumps capable of injecting into the RCS.~~ ~~OR~~ ~~One charging pump and one safety injection pump capable of injecting into the RCS.~~ ~~OR~~ ~~Two or more safety injection pumps capable of injecting into the RCS.~~		~~NOTE~~ ~~Two charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes.~~ ---------------------------------	
		A.1	Initiate action to verify a maximum of ~~one charging~~ Two pumps ~~or one safety injection pump is~~ are capable of injecting into the RCS.	Immediately
B.	Reactor coolant pump operation not limited as specified in Table 3.4.12-1.	B.1	Initiate action to limit pump operation as specified in Table 3.4.12-1.	Immediately
C.	An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in Specification 3.4.3.	C.1	Isolate affected accumulator.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > ~~285~~ 210°F. OR	12 hours
	D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed by Specification 3.4.3.	12 hours
E. One ~~PORV~~ required RCS relief valve inoperable in MODE 4.	E.1 Restore ~~PORV~~ required RCS relief valve to OPERABLE status.	7 days
F. One ~~PORV~~ required RCS relief valve inoperable in MODE 5 or 6.	F.1 Restore ~~PORV~~ required RCS relief valve to OPERABLE status.	24 hours
G. Two ~~PORVs~~ required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, D, E, or F, not met. OR LTOP System inoperable for any reason other than Condition A, C, D, E, or F.	G.1 Initiate ACTION to ensure a maximum of one charging pump or one safety injection pump is capable of injecting into the RCS AND G.~~1~~2 Depressurize RCS and establish RCS vent of ≥ 4.5 square inches.	Immediately ~~8~~ 12 hours

two pumps (charging, safety injection, or charging and safety injection) are

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of ~~one charging pump or safety injection pump is~~ capable of injecting into the RCS.	12 hours
SR 3.4.12.2	Verify each accumulator is isolated.	12 hours
SR 3.4.12.3	-----------------------------NOTE----------------------------- Only required to be performed when complying with ~~LCO 3.4.12.b.~~ REQUIRED ACTION G.2 -- Verify required RCS vent ≥ 4.5 square inches open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)
SR 3.4.12.4	Verify PORV block valve is open for each required PORV.	72 hours
SR 3.4.12.5	-----------------------------NOTE----------------------------- Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ ~~285~~ 210°F. -- Perform a COT on each required PORV, excluding actuation.	31 days
SR 3.4.12.6	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	18 months
SR 3.4.12.7	Verify both associated RHR suction isolation valves are open with operator power removed for each required RHR suction relief valve.	12 hours

Table 3.4.12-1 (Page 1 of 1)

(UNIT 1 ONLY)

Reactor Coolant Pump Operating Restrictions for Low Temperature Overpressure Protection

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
~~≥ 67°F~~	~~1~~
≥ ~~68~~ 70°F	2
≥ ~~78~~ 126°F	4

Table 3.4.12-1 (Page 1 of 1)

(UNIT 2 ONLY)

Reactor Coolant Pump Operating Restrictions for Low Temperature Overpressure Protection

Reactor Coolant System Cold Leg Temperature	Maximum Number of Pumps Allowed in Operation
≥ ~~67~~°F 70	1
~~≥ 73°F~~	~~2~~
≥ ~~95~~°F 140	4

A second program that employs excore cavity dosimetry to monitor the reactor vessel neutron fluence will be installed in each unit. This program will meet the requirements of 10 CFR 50 Appendix H.

BASES

BACKGROUND (continued)

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

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

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

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

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 7).

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

BASES

LCO (continued)

performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ ~~285~~ 210°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2. The water level is maintained by an OPERABLE AFW train in accordance with LCO 3.7.5, "Auxiliary Feedwater System."

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side narrow range water level ≥ 12%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side narrow range water levels ≥ 12%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ ~~285~~°F.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a locked rotor. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ ~~285~~ 210°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper pressure limit of +3% is consistent with the ASME requirement (Ref. 1) for lifting pressures above 1000 psig. The lower pressure limit of -2% is selected such that the minimum LCO lift pressure remains above the uncertainty adjusted high pressure reactor trip setpoint. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of

BASES

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ ~~285~~ 210°F or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

BASES

ACTIONS (continued)

B.1 and B.2

210

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ ~~285~~°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below ~~285~~°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is +3% and -2% of the nominal setpoint of 2485 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Chapter 15.
3. UFSAR, Section 5.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

BASES

APPLICABILITY (continued)

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for 210 LTOP in MODES 4 ≤ ~~285~~°F, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.

A.1

With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. This specification provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

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

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

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. two pumps Limiting coolant input capability requires all but ~~one charging pump or one safety injection pump~~ incapable of injection into the RCS, isolating the accumulators, and limiting reactor coolant pump operation at low temperatures. The pressure relief capacity requires ~~either~~ two redundant RCS relief valves ~~PORVs or a depressurized RCS and an RCS vent of sufficient size~~. One RCS relief valve ~~PORV or the open RCS vent~~ is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve

BACKGROUND (continued)

decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one charging pump for makeup in the event of loss of inventory, then additional pumps can be made available through manual actions.

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

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure reaches 400 psig (as left calibrated), allowable value ≤ 425 psig (as found), when the PORVS are in the "lo-press" mode of operation. If the PORVs are being used to meet the requirements of this Specification, then indicated RCS cold leg temperature is limited to ≥ ~~65~~ 70°F in accordance with the LTOP analysis. When all Reactor Coolant Pumps are secured, this temperature is measured at the outlet of the residual heat removal heat exchanger. This location will provide the most conservative (lower) temperature measurement of water capable of being delivered into the Reactor Coolant System. The LTOP actuation logic monitors both RCS temperature and RCS pressure. The signals used to generate the pressure setpoints originate from the wide range pressure transmitters. The signals used to generate the temperature permissives originate from the wide range RTDs. Each signal is input to the appropriate NSSS protection system cabinet where it is converted to an internal signal and then input to a comparator to generate an actuation signal. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

This Specification presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits ensures that the Reference 1 limits will not be exceeded in any analyzed event.

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

BASES

BACKGROUND (continued)

Insert 1

~~RCS Vent Requirements~~

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

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 3) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding ~~285~~ 210°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about ~~285~~ 210°F and below, overpressure prevention falls to two OPERABLE ~~PORVs or to a depressurized RCS and a sufficient sized RCS vent~~ RCS relief valves. Each of these means has a limited overpressure relief capability.

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

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

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

Mass Input Type Transients

a. Inadvertent safety injection ~~;~~ of one safety injection pump and one charging pump or

b. Charging/letdown flow mismatch.

INSERT 1

RHR Suction Relief Valve Requirements

During LTOP MODES, the RHR system is operated for decay heat removal and low-pressure letdown control. Therefore, the RHR suction isolation valves (there are two suction isolation valves per line) are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open with operator power removed to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows type water relief valve with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 8) for Class 2 relief valves.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

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

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

a. Rendering all but two ~~one charging pump or one safety injection~~ pumps incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions;

c. Limiting RCP operation based on the existing temperature in the RCS cold legs; and

d. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops—MODE 4," and LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," provide this protection.

RCS Relief valve

The Reference 3 analyses demonstrate that ~~either~~ one ~~PORV or the depressurized RCS and RCS vent~~ can maintain RCS pressure below limits when ~~only one charging pump or one safety injection pump is~~ actuated. Thus, the LCO allows ~~only one charging pump or one safety injection~~ pumps OPERABLE during the LTOP MODES. ~~Since neither one PORV nor the RCS vent can handle the pressure transient from accumulator injection when RCS temperature is low,~~ The LCO also requires the accumulators ~~isolation~~ when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3.

any two pumps (charging and/or safety injection) are

two

be isolated

The isolated accumulators must have their discharge valves closed and power removed.

The restrictions on the number of RCPs in operation at a given temperature ensures that during a LTOP mass injection event that the pressure/temperature (P/T) limits of 10 CFR 50, Appendix G to protect the

BASES

APPLICABLE SAFETY ANALYSES (continued)

reactor vessel are not exceeded. During startup and shutdown, when the RCPs are operated, their induced flows create a pressure drop across the vessel. This pressure drop along with the difference in elevation between the beltline region and the instrumentation locations are additive to the peak pressure from the mass injection event.

The amount of the pressure at the reactor vessel beltline region from the RCPs is dependent on the number of RCPs operated. Adequate margin to prevent exceeding the P/T limits is assured by restricting the number of RCPs operated. Since LTOP events are basically acknowledged as being steady-state events, these RCP operating restrictions are designed to work with the LTOP setpoint to provide protection from exceeding the steady-state Appendix G P/T limits.

Fracture mechanics analyses established the temperature of LTOP Applicability at ~~285~~°F. 210

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

two pumps (charging and/or safety injection)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the specified limit. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one charging pump ~~or~~ and one safety injection pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

~~RCS Vent Performance~~

Insert 2

~~With the RCS depressurized, analyses show a vent size of 4.5 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one charging pump or one safety injection pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.~~

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

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

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6).

LCO

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

two pumps (charging and/or safety injection)

To limit the coolant input capability, the LCO permits a maximum of ~~one charging pump or one safety injection pump~~ capable of injecting into the RCS and requires all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3. The LCO also limits RCP operation based on existing RCS cold leg temperature as required by the LTOP analysis.

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

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit.

Insert 3

INSERT 2

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses show that one RHR suction relief valve with a setpoint at or between 417 psig and 509 psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation $\leq$ 10% of the rated lift setpoint.

Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR system does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.

The RHR suction relief valves are considered to be active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.

INSERT 3

b. Two OPERABLE RHR suction relief valves (ND-3 and ND-38); or

   An RHR suction relief valve is OPERABLE for LTOP when both of its RHR suction isolation valves are open, its setpoint is at or between 417 psig and 509 psig, and testing has proven its ability to open in this pressure range.

c. One OPERABLE PORV and one OPERABLE RHR suction relief valve.

BASES

LCO (continued)

~~b. A depressurized RCS and an RCS vent.~~

~~An RCS vent is OPERABLE when open with an area of ≥ 4.5 square inches.~~

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

APPLICABILITY

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

210

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

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

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

LCO 3.0.4 is not applicable for entry into LTOP operation.

A.1

With More than two ~~or more charging~~ pumps, (Charging and/or safety injection ~~two safety injection pumps, or one safety injection pump and one charging pump~~ capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

BASES

ACTIONS (continued)

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

B.1

With RCP operation not limited in accordance with Table 3.4.12-1, RCS overpressurization is possible.

To immediately initiate action to limit pump operation reflects the urgency of removing the RCS from this condition.

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

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > ~~285~~°F, an accumulator pressure of 678 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit also gives this protection.

210

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

RCS relief valve

E.1

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

RCS relief valves (in any combination of the PORVs and RHR Section relief valves)

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

RCS relief valves

BASES

ACTIONS (continued)

F.1

RCS relief valves

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two ~~PORVs~~ inoperable in MODE 5 or in MODE 6 with the head on Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE ~~PORV~~ to protect against overpressure events.

RCS relief valve

G.1

Steps must be taken immediately to limit potential mass input into the RCS and

t ~~The RCS must be depressurized and a vent must be established within~~ 12 ~~8~~ hours when:

required RCS relief valves

a. Both ~~PORVs~~ are inoperable; or

b. A Required Action and associated Completion Time of Condition A, D, E, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, C, D, E, or F.

Insert 4

~~The vent must be sized ≥ 4.5 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.~~

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, ~~all but one charging or safety injection pump is verified incapable~~ of injecting into the RCS and the accumulator discharge isolation valves are verified closed and power removed.

a maximum of two pumps (charging and/or safety injection) are verified capable

INSERT 4

The Reference 3 analyses demonstrate that with the mass input into the RCS reduced to that of one injection pump (charging or safety injection) an RCS vent of $\geq$ 4.5 square inches can maintain RCS pressure below limits. Therefore the Condition requires action to be taken immediately to reduce the input to that on one injection pump (charging or safety injection) prior to commencing RCS pressure reduction and establishing the required RCS vent. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle fracture of the reactor vessel.

The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one charging pump or one safety injection pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. The required vent capacity may be provided by one or more vent paths. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

The pumps are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through two valves in the discharge flow path being closed.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

(valves that are sealed or secured in the open position, are considered "locked" in this context); or

SR 3.4.12.3

The RCS vent of ≥ 4.5 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not ~~cannot be~~ locked.

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position or a removed pressurizer safety valve or open manway also fits this category). ~~a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.~~

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the REQUIRED ACTION G.2 ~~LCO 3.4.12b~~.

SR 3.4.12.4

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

BASES

SURVEILLANCE REQUIREMENTS (continued)

210

SR 3.4.12.5

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ ~~285~~°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required.

The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.

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

SR 3.4.12.6

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

Insert 5

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. UFSAR, Section 5.2
4. 10 CFR 50, Section 50.46.
5. 10 CFR 50, Appendix K.
6. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section III
9. ASME, Boiler and Pressure Vessel Code, Section XI

INSERT 5

SR 3.4.12.7

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open with operator power removed and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.

The RHR suction isolation valves are verified to be opened with operator power removed every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 9), test per Inservice Testing Program verifies OPERABILITY by proving relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

ATTACHMENT 2

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

DESCRIPTION OF PROPOSED CHANGES AND TECHNICAL JUSTIFICATION

Background:

Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits

Technical Specification (TS) Bases 3.4.3 includes the background regarding the RCS P/T limits. This specification contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature derived using the AMSE Boiler and Pressure Vessel Code Section XI. Each P/T limit curve defines an acceptable region for normal operation. Currently, Catawba Nuclear Station (CNS) Units 1 and 2 P/T limits have been evaluated for up to 15 effective full power years (EFPY). This amendment request provides the justification for the new P/T limits. These changes rely in part on an alternative methodology used in determining allowable P/T limits (ASME Code Case N-640) and evaluation of the latest irradiated reactor vessel material specimens. Duke requested Westinghouse to perform reactor vessel integrity assessments and generate new P/T limit curves for Units 1 and 2. These curves have been developed and envelop operation up to 34 EFPY for both units as detailed in Westinghouse Reports WCAP-15203 and WCAP-15285 (see Attachments 7 and 8).

RCS Low Temperature Overpressure Protection (LTOP) System

TS Bases 3.4.12 included the background regarding the LTOP system. The LTOP pressure and temperature setpoints provide restrictions for the protection from non-ductile failure of the RCS under transient conditions. The LTOP system protects the reactor vessel from excessive pressures at low temperature condition. LTOP calculations provide inputs to or verification of the LTOP system and associated TS. Each time the P/T curves are revised, the LTOP system must be re-evaluated to ensure its functional requirement can still be met using the RCS power operated relief valve (PORV) method or depressurized and vented RCS method.

The proposed P/T limit curves and LTOP setpoints satisfy the requirement of 10 CFR 50.60(a) with two exceptions. The first exception is the use of ASME Boiler and Pressure Vessel Code Case N-640. The second is the use of ASME Code Case N-641 in determining the LTOP system enable temperature. The justifications for these exceptions are included in Attachments 5 and 6.

ASME Code Case N-640

The startup and shutdown process for an operating nuclear plant is controlled by P/T limit curves, which are developed, based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI and incorporate four numbers of safety margins, one of which is the lower bound fracture toughness curve. There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is the lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1," allows the use of K_{IC} fracture toughness curve instead of K_{IA} fracture toughness curve for the development of P/T limit curves. The other margins involved with the process remain unchanged. This code case was used in the development of the P/T limit curves and documented in Westinghouse Reports WCAP-15203 and WCAP-15285 (see Attachments 7 and 8).

Duke requests an exemption to use ASME Code Case N-640 pursuant to 10 CFR 50.60(b) and 10 CFR 50.12. This exemption request is provided in Attachment 5 along with the Code Case N-640 and accompanying technical basis. This attachment contains a list of NRC-Approved industry exemptions and amendments related to Code Case N-640 application and P/T limit changes.

ASME Code Case N-641

ASME Code Case N-641 permits utilizing methodology for the formulation of an enable temperature that maintains the margin of safety inherent in the generic formulation for the enable temperature. Application of Code Case N-641 permits the implementation of an LTOP enable temperature that preserves an acceptable margin of safety while maintaining operational margins for reactor coolant pump operation at low temperatures and pressures. The LTOP system enable temperature established in accordance with ASME Code Case N-641 will also minimize the unnecessary actuation of protection system pressure relieving devices. Therefore, establishing the LTOP enable temperature in accordance with ASME Code Case N-641 satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety. Duke requests exemption to use ASME Code Case N-641 pursuant to 10 CFR 50.60(b) and 10 CFR 50.12. This exemption request is provided in Attachment 6 along with the Code Case N-641 and accompanying technical basis. This attachment contains a list of NRC-Approved industry exemptions and amendments related to Code Case N-641 application.

Residual Heat Removal (RHR) System

The design of the RHR system includes two motor-operated gate isolation valves in series on each inlet line between the high pressures RCS and the lower pressure RHR system. They are closed during normal operation and are only opened for residual heat removal during a unit cooldown after the RCS pressure is reduced to approximately 385 psig and RCS temperature is reduced to approximately 350°F. During a unit startup the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above approximately 385 psig. These isolation valves are provided with "prevent-open" interlocks which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure. The two inlet isolation valves in each subsystem are separately and independently interlocked with pressure signals to prevent their being opened whenever the RCS pressure is greater than approximately 385 psig. Each inlet line to the RHR system is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief valve set pressure. These relief valves also protect the RHR system from inadvertent overpressurization during plant cooldown or heatup.

Description of Proposed Changes

The following changes to the TS are proposed:

1. TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits," are revised to reflect the new P/T limits that are effective to a maximum of 34 EFPY for Units 1 and 2. Figures 3.4.3-1(Unit 1 only), 3.4.3-1(Unit 2 only), 3.4.3-2(Unit 1 only) and 3.4.3-2(Unit 2 only) are replaced with revised figures for operation up to 34 EFPY. TS Bases 3.4.3 is revised to include the excore cavity dosimetry program to be installed. The reactor vessel (RV) in-core surveillance capsule program requirements are expected to be completed during EOC 14 which is projected during the fall of 2003 for Catawba Unit 1 and spring of 2006 for Catawba Unit 2. The station will continue RV fluence monitoring through a second program that employs excore cavity dosimetry to determine the RV fluence through calculation-based fluence determination. Both Catawba units will install excore cavity dosimetry during a future refueling outage for each unit.

2. TS 3.4.6, "RCS Loops - MODE 4," is revised to reflect the change in the LTOP system enable temperature from 285°F to 210°F. The associated TS Bases are also revised to reflect this change.

3. TS 3.4.7, "RCS Loops - MODE 5, Loops Filled," is revised to reflect the change in the LTOP system enable temperature from 285°F to 210°F. The associated TS Bases are also revised to reflect this change.

4. TS 3.4.10, "Pressurizer Safety Valves," is revised to reflect the change in the LTOP system enable temperature from 285°F to 210°F. The associated TS Bases are also revised to reflect this change.

5. The Bases for TS 3.4.11, "Pressurizer Power Relief Valves (PORVs)," are revised to reflect the change in the LTOP system enable temperature from 285°F to 210°F.

6. Several changes are being made to TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." They are as follows:

   a) The Limiting Condition for Operation (LCO) is being revised to allow a maximum combination of two pumps (charging pumps or safety injection pumps or charging and safety injection pumps) capable of injecting into the RCS. In support of this change, Condition A is revised to reflect the changes in the LCO for two pumps capable of injecting into the RCS. The note prior to Required Action A.1 is deleted since it is no longer applicable. Required Action A.1 is revised to reflect the changes to Condition A. Surveillance Requirement (SR) 3.4.12.1 is also revised to verify that a maximum of two pumps are capable of injecting into the RCS.

   b) The LCO is also revised to reflect that the residual heat removal (RHR) suction relief valves are acceptable to be used as an RCS vent path during LTOP system operation. The LTOP calculations have been revised and determined that the RHR suction relief valves are adequate for LTOP operation. In support of this change, Conditions E, F, and G are being revised to address the additional relief capabilities as described above. Surveillance Requirement (SR) 3.4.12.3 is revised to add the word "required" prior to RCS vent. This was done to clarify that the SR only applies to the RCS vent paths that are required to meet the LCO. One SR is added as proposed SR 3.4.12.7 that states "Verify both associated RHR suction isolation valves are open with operator power removed for each required RHR suction relief valve." The frequency of this SR is every 12 hours. This revision is consistent with the Improved Standard Technical

Specifications for Westinghouse Reactors located in NUREG-1431, Revision 2. Some changes from the Improved Standard Technical Specifications are made due to plant specific terminology to eliminate any potential confusion.

c) The LCO is also being revised to remove the allowance for a depressurized RCS with an RCS vent of $\geq$ 4.5 square inches. This allowance is no longer applicable for the case where two pumps are capable of injection into the RCS when LTOPs is required by TS. This allowance has been relocated to revised Required Action G, which requires a maximum of one pump capable of injecting into the RCS whenever, an RCS vent of $\geq$ 4.5 square inches is established.

d) A new Required Action G.1 has been added to ensure that if two required RCS relief valves are not operable, immediate action is taken to ensure that a maximum of one pump is capable of injecting into the RCS. The previous Required Action G.1 is changed to G.2 and the completion time is changed from 8 hours to 12 hours. This change provides operator flexibility for a more controlled depressurization and establishment of an RCS vent. The slight extension in the completion time does not represent a significant risk increase during shutdown operations. These changes are consistent with NUREG-1431, revision 2.

e) The TS is also revised to reflect the change in the LTOP system enable temperature from 285°F to 210°F throughout the TS.

f) Table 3.4.12-1, Reactor Coolant Pump Operating Restrictions for Low Temperature Overpressure Protection," for Unit 1 ONLY and Unit 2 ONLY are revised to reflect the limits on reactor coolant pump operation when LTOP is inservice based on the revised heatup and cooldown curves.

g) Two references to the ASME Boiler and Pressure Vessel Code were added to reflect changes to the TS Bases for the RHR suction relief valves.

h) The TS 3.4.12 Bases are revised to reflect the changes described above.

Technical Justification

Pressure-Temperature Limits

Determination of Adjusted RT_{NDT} (ART)

The projected 34 EFPY ART values at the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations for the beltline regions of the Catawba reactor vessels were calculated by Westinghouse. These calculations were in accordance with Regulatory Guide 1.99, Revision 2. Regulatory Guide 1.99, Revision 2 credibility criteria are applied by Westinghouse to determine the appropriate margin term. The calculations determined the ART for the various reactor vessel (RV) materials using Regulatory Guide 1.99, Revision 2, Regulatory Positions 1.1 and 2.1. The selected controlling values are those RV locations with the highest ART for 1/4T and 3/4T whether determined using Regulatory Position 1.1 or 2.1 methodologies.

The calculation of the ART values for the 1/4T and 3/4T locations at 34 EFPY for Unit 1 is presented in WCAP-15203, Table 8 and Table 9 (reference 1). As indicated by these tables, the limiting ART values used in the generation of the Unit 1 P/T curves are 42°F at the 1/4T location and 31°F at the 3/4T location. The limiting material for Unit 1 was determined to be the intermediate and lower shell forging 05 & 04.

The calculation of the ART values for the 1/4T and 3/4T locations at 34 EFPY for Unit 2 is presented in WCAP-15285, Table 10 and Table 11 (reference 2). As indicated by these tables, the limiting ART values used in the generation of the Unit 2 P/T curves are 121°F at the 1/4T location and 106°F at the 3/4T location. The limiting material for Unit 2 was determined to be the intermediate shell plate B8605-2.

Westinghouse conservatively provided 100 % of the steady state Appendix G limits applying Code Case N-640 for both units. Since appropriate instrument error allowances are included in the operating procedures, the Technical Specification P/T limit curves do not include margins for instrument error.

Determination of Pressure-Temperature Limits

The proposed P/T limits for Units 1 and 2 were developed using Westinghouse computer code OPERLIM, as modified by ASME Code Case N-640 for use of the K_{IC} fracture toughness curve. The methods and criteria employed to establish operating pressure and temperature limits are described in NRC-approved Report WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold

Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves." The method of analysis consists of determining the P/T limits for the beltline region including the closure head flange region of the reactor vessel for normal heatup, normal cooldown, and inservice leak and hydrostatic test. The P/T limit curves and supporting technical basis generated by Westinghouse are included in the WCAPs in Attachments 7 and 8 of this amendment.

Reactor Closure Head/Vessel Flange Requirements

Westinghouse Reports WCAP-15203 for Unit 1 (Attachment 7) and WCAP-15285 for Unit 2 (Attachment 8) used a method that was less conservative than the 10 CFR 50 Appendix G method for calculating the closure head / vessel flange requirements. This alternate methodology is similar to that in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants." WCAP-15315 has not been approved by the NRC. The NRC stated in a letter dated August 1, 2001 to the McGuire Nuclear Station that the NRC staff does not plan to review any exemption requests until the staff completes the review of WCAP-15315. Therefore, CNS has performed calculations (reference 5) to include the requirements in 10 CFR 50 Appendix G for the heatup and cooldown curves for both Unit 1 and Unit 2.

10 CFR Part 50, Appendix G, addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure head regions must exceed the material unirradiated RT_{NDT} by at least 129°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure. With a pre-service hydrostatic test pressure of 3107 psig, the Appendix G limit is 621 psig for the Catawba Units 1 and 2 reactor vessels.

For Unit 1, the limiting unirradiated RT_{NDT} of -4 °F occurs in the closure head / vessel flange region. The minimum allowable temperature of this region is 116 °F [T = 120 °F + (-4 °F)] at pressures greater than 621 psig, with no margins for instrument uncertainties.

For Unit 2, the limiting unirradiated RT_{NDT} of 10 °F occurs in the closure head / vessel flanges region. The minimum allowable temperature of this region is 130 °F [T = 120 °F + 10 °F] at pressures greater than 621 psig, with no margins for instrument uncertainties.

Excore Cavity Dosimetry Program

TS 3.4.3 Bases is revised to briefly describe a second program to monitor the reactor vessel neutron fluence that is currently scheduled to be installed in Unit 1 during 1EOC14 and in Unit 2 during 2EOC13. The new program employs excore cavity dosimetry to monitor and determine RV neutron fluence within a limited amount of uncertainty through calculation-based fluence determination. Cavity dosimetry calculations are aligned to meet the intentions given in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001.

Cavity dosimetry measurements are used to verify the accuracy of fluence calculations and to determine fluence uncertainty values. Dosimetry removed from the cavity will be laboratory tested to evaluate material radiation effects. Computer analyses calculate accumulated fast neutron fluence using these laboratory measurements. Westinghouse employs a calculation-based fluence analysis methodology that can be used to predict the fast neutron fluence in the RV using cavity dosimetry to benchmark the fluence predictions. Regulatory Guide 1.190 specifies that the results of the fluence analysis are expected to be within 20% of the calculated values.

LTOP Enable Temperature

The LTOP enable temperature was calculated using the method describe in ASME Code Case N-641, Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1. This Code Case provides the following in paragraph 2215.2a, "the LTOP system effective temperature T_{enable} is the temperature at or above which the safety relief valves provide adequate protection against non-ductile failure." LTOP systems shall be effective below the higher temperature calculated utilizing the methods described in the ASME Code Case N-641.

This results in a Catawba Unit 1 LTOP Enable Temperature of 210°F for the new Technical Specification data that would be valid up to 34 EFPY.

This results in a Catawba Unit 2 LTOP Enable Temperature of 210°F for the new Technical Specification data that would be valid up to 34 EFPY.

LTOP Operation with a Maximum of Two Pumps Capable of Injection and Residual Heat Removal Suction Relief Valves

It is desirable to operate with combinations of both charging pump and safety injection pumps in service for brief periods during plant heatup (i.e. for accumulator fill and check valve testing). It is also desirable to take credit for the Residual Heat Removal (RHR) suction relief valves capacity in addition to the Pressurizer PORVs for compliance with LTOP Technical Specification. Doing so would provide operational, maintenance, and test flexibility for more efficient outage planning and would improve response time for a MODE 4 LOCA. This can be shown to be acceptable provided the RHR system suction relief valve is available to relieve the required capacity. The following description summarizes the evaluation of the adequacy of the RHR suction relief valves and PORVs to provide RCS relief protection in this configuration.

The PORV setpoint is verified acceptable by comparison to the Pressure/Temperature (P/T) curves developed by WCAP-15203 (reference 1) and WCAP-15285 (reference 2). Acceptable heatup and cooldown limits are determined based on comparison of peak pressure and P/T curve limits. LTOP protection when using the RHR suction relief valves is verified in a manner similar to the method used to verify an adequate PORV setpoint. The nominal relief valve setpoint is adjusted based on various uncertainties to arrive at a peak relieving pressure for the limiting mass input transient.

Flow values for the limiting mass input transient are calculated using conservatively adjusted minimum performance curves (head vs. flow) for safety injection and charging pumps and a system curve calculated based on actual test data. The safety injection and charging pump flow rates are calculated based on a conservative set of operating parameters. The flow rates selected for LTOP analysis are based on the flow rate of the specific pump at the LTOP relief valve setpoint. The results of the charging pump and safety injection pump flow analysis is as follows:

Charging and Safety Injection Pump System Flow Analysis (400 psi Setpoint)	
Pump / Combination	Flow (gpm)
1 charging pump	475
2 charging pumps	660
1 safety injection pump	550
2 safety injection pumps	690
1 charging pump + 1 safety injection pump	475 + 550 = 1025

The peak pressure resulting from a 400-psig PORV setpoint was calculated in CNC-1223.03-00-005 (reference 6). The limiting transient was the injection of a charging pump and safety injection pump. The peak pressure was determined to be 685.7 psig. The minimum pressure at steady state conditions was determined to be 719 psig. Since the peak pressures is less than the most limiting pressure this PORV setpoint and capacity is sufficient to protect the reactor coolant system from cold overpressure events. It follows that this setpoint is also adequate for all the other combinations of operating injection pumps analyzed in the calculation. Because the peak pressure result from a single PORV is below the Appendix G limit, the revised LTOP TS requires two RCS relief valves to be operable (assuming for an active single failure) to mitigate the consequences of any pressure transient.

The characteristics of the RHR suction relief valves are a cold set pressure setpoint of 463 psig with a set point tolerance of +/- 10% for relief valve setting drift and a capacity of 2027 gpm. These values were used to calculate the peak pressure when the relief valves were relieving at maximum capacity. The peak pressures varied depending upon the number of reactor coolant pumps that were assumed to be operating. These numbers were compared to the allowable heatup and cooldown limits developed for 34 EFPY. The allowable pressure for closure head/vessel flange region is 621 psig. The peak pressure calculated for the RHR suction relief valves was 602.3 psig with a maximum of two (2) reactor coolant pumps operating. Utilizing WCAP 15203 and WCAP 15285 the peak pressure of 602.3 psig determines a limiting temperature of 60°F for both units. Therefore, for both units, the RHR relief valves are adequate for all steady state conditions with 2 reactor coolant pumps operating at RCS temperatures ≥ 60°F (≥ 70°F with instrument uncertainty). However, to prevent exceeding the 621 psig limit with the PORVs providing overpressure protection, the number of reactor coolant pumps operating will be restricted to 2 reactor coolant pumps on Unit 1 and 1 reactor coolant pump on Unit 2 at reactor coolant system temperatures ≥ 60°F (≥ 70°F with instrument uncertainty).

The calculations determined a peak pressure of 660.0 psig (peak pressure for RHR relief valves with four (4) reactor coolant pumps operating). The steady state values will be referenced at temperatures above closure head/vessel flange region limits. Therefore, for both units, the RHR relief valves are adequate for all steady state conditions with 4 reactor coolant pumps operating at Unit 1 RCS temperatures ≥ 116°F (≥ 126°F with instrument uncertainty) and Unit 2 RCS temperatures ≥ 130°F (≥ 140°F with instrument uncertainty.) These reactor coolant pump operating restrictions are bounded by the reactor coolant pump operation restrictions applicable to the pressurizer PORVs, and therefore, impose no additional restriction on operation of the LTOP system. TS Table 3.4.12-1 for each unit has been revised to reflect this requirement.

In order to develop the flexibility to operate with two pumps capable of injecting into the RCS during LTOPs some operational flexibility required modification. The modification was the ability to take credit for the RCS depressurized with an RCS vent of $\geq$ 4.5 square inches to meet the LCO was removed. This ability was left in Required Action G.2 provided only one pump was capable of injecting into the RCS. Calculations for the two-pump case for an RCS vent of this size were not done due to its complexity and that calculations demonstrated that both the PORVs and RHR relief valves provided the relief capacity and redundancy for routine operations.

The calculation results have shown that a single PORV or a single RHR suction relief valve is adequate to mitigate the pressure increase from the worst case transient (mass input from one safety injection pump + one charging pump).

Minimum Temperature in LTOP Mode

For the test capsules analyzed in 1999, Duke Power contracted Westinghouse to generate new heatup and cooldown curves using K_{IC} in place of K_{IR} for the calculation of the stress intensity factors. The heatup and cooldown (P-T) curves were generated without margins for instrumentation errors and included a hydrostatic leak test limit curve from 2485 to 2000 psig and pressure - temperature limits for the reactor vessel flange regions per the requirements of 10 CFR 50, Appendix-G, as described in the reports. The P-T curves in the Westinghouse Reports (WCAP-15203 reference 1 and WCAP-15285 reference 2) were developed with the identical adjusted reference temperature (ART) values previously used.

This information was used in Catawba calculation CNC-1223.03-00-005 (reference 6) to determine the minimum temperature for LTOP. The minimum temperature allowed while in LTOP mode is found using the peak pressure from the different events analyzed in reference 5 and comparing this value to the steady state pressure / temperature values from Table 2.A and 2.B of reference 6. This determined that the minimum temperature for LTOP mode is 70°F. This temperature limit includes a 10°F temperature uncertainty.

Summary of Technical Analysis

The proposed changes to the P/T and LTOP limits satisfy the requirements of 10 CFR 50 Appendix G, Appendix H, and ASME Section XI Appendix G, as modified by Code Case N-640 and N-641. The calculation of ART is consistent with the method in RG 1.99, Revision 2. The calculation of fluence values is consistent with the guidance in RG 1.190. The LTOP changes are performed in accordance with approved procedures under Duke QA program and are consistent with the method in ASME Code Case N-640 and Code Case N-641. Duke concludes that the proposed changes conform to the underlying purpose of NRC's regulations and maintain the safe operation of the station.

References

1. WCAP-15203, Catawba Unit 1 Heatup and Cooldown Curves for Normal Operation Using Code Case N-640, Revision 1, April 2001.
2. WCAP-15285, Catawba Unit 2 Heatup and Cooldown Curves for Normal Operation Using Code Case N-640, October 1999.
3. ASME Code Case N-640, Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1.
4. ASME Code Case N-641, Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1.
5. CNC-1223.03-00-005, Pressurizer Power Operated Relief Valve Setpoint Verification for Low Temperature Overpressure Protection, January 19, 2003.
6. Pressure Mitigating Systems Transient Analysis Results, Westinghouse Electric Corporation for Reactor Coolant Overpressurization, July 1977.
7. Supplement to the July 1977 Report; Pressure Mitigating Systems Transient Analysis Results, Westinghouse Electric Corporation for the Reactor Coolant System Overpressurization, September 1977.
8. CNC-1210.04-00-0064, Loop Accuracy Calculation for LTOP and Pressure Setpoints. Revision 1, June 1997.

Industry Exemptions and Amendments Related to Code Case N-640

Plant Name	Application Date	Exemption Date	Amendment Date
Oconee	5/11/99	7/23/99	10/1/99
Beaver Valley 2	6/17/99	9/06/00	9/06/00
VC Summer	8/19/99	10/20/99	10/21/99
Dresden	2/23/00	8/25/00	9/19/00
Shearon Harris	4/12/00	7/26/00	7/28/00
Limerick	5/15/00	9/07/00	9/15/00
Hatch	6/01/00	8/29/00	8/29/00
Clinton	8/25/00	10/30/00	10/31/00
Calvert Cliffs	9/14/00	2/26/01	3/15/01
Vermont Yankee	12/19/00	4/16/01	5/04/01

Industry Exemptions and Amendments Related to Code Case N-641

Plant Name	Application Date	Exemption Date	Amendment Date
North Anna	6/22/00	5/2/01	5/2/01
Turkey Point	7/7/00	10/24/00	10/24/00
Point Beach	7/14/00	10/6/00	8/8/01
ANO-2	10/30/01	04/15/02	04/15/02

ATTACHMENT 3

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

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

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

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

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the reactor coolant system (RCS) pressure-temperature (P/T) limits are developed utilizing the methodology of ASME XI, 10 CFR 50 Appendix G, in conjunction with the methodology of Code Case N-640. Usage of these methodologies provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides operational limits that ensure failure of the reactor vessel will not occur. The proposed changes to allow operation with two pumps capable of injecting into the RCS and utilization of the residual heat removal (RHR) suction relief valves has been evaluated and determined to be provide adequate protection of the RCS from the worst case pressure transient.

The probability of any design basis accident (DBA) is not affected by these changes, nor are the consequences of any DBA affected by these changes. The P/T limits, and low temperature overpressure protection (LTOP) setpoints, and T_{enable} value are not considered to be initiators or contributors to any accident analysis addressed in the Catawba UFSAR.

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. 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. The proposed changes to the TS are consistent with the intent of the flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 2.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the updated final safety analysis report (UFSAR) because the accident analysis assumptions and initial conditions will continue to be maintained.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not involve any physical alteration of plant systems, structures, or components. The requirements for the P/T limit curves and LTOP setpoints remain in place. The fundamental approach follows approved ASME and Westinghouse report methodology. The proposed curves and change to the enable temperature for LTOP system reflect changes in material properties acknowledged and managed by regulation and an upgrade in technology, which has been approved by ASME.

The proposed changes to allow operation with two pumps capable of injecting into the RCS and utilization of the RHR suction relief valves has been evaluated. The evaluation has shown that both the PORVs and RHR suction relief valves provide adequate relief protection of the RCS from the worst case pressure transient and provide equivalent protection to that already allowed by the current TS.

The proposed changes do not introduce new failure mechanisms for system structures, or components not already considered in the UFSAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created because no new failure mechanisms or initiating events have been introduced.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed changes are developed utilizing the methodology of ASME XI, 10 CFR 50 Appendix G, in conjunction with Code Case N-640 and Code Case N-641 methodology. Usage of these methodologies provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides operational limits that ensure failure of the reactor vessel will not occur. Although the Code Cases constitute relaxation from the current requirements of 10 CFR 50 Appendix G, the alternative methodology allowed by the Code is based on industry experience gained since the inception of the 10 CFR 50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Cases maintain a margin of safety that is consistent with the intent of 10 CFR 50 Appendix G, i.e., with regard to the margin originally contemplated by 10 CFR 50 Appendix G for determination of RCS P/T limits.

The analyses completed for this proposed TS amendment demonstrate that established acceptance criteria continue to be met. Specifically, the P/T limit curves, LTOP setpoints, allowances for operating two pumps, utilization of RHR suction relief valves and LTOP T_{enable} values provide acceptable margin to vessel fracture under both normal operation and LTOPs design basis (mass addition and heat addition) accident conditions. The proposed changes to the TS are consistent with the intent of the flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 2. Therefore, there will be no significant reduction in a margin of safety as a result of the proposed changes.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

ATTACHMENT 4

ENVIRONMENTAL ANALYSIS

Environmental Analysis

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of this amendment will have no adverse impact upon the Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Catawba TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

ATTACHMENT 5

Justification for ASME Code Case N-640 Exemption Request

Justification for ASME Code Case N-640 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-640, "Alternative Fracture Toughness for Development of Pressure-Temperature Limit Curves for ASME Section XI, Division 1" in lieu of 10 CFR 50, Appendix G.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow the use of ASME Code Case N-640 in conjunction with ASME Section XI, Appendix G to determine the Pressure-Temperature (P/T) limits meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states the Commission may grant an exemption from the requirements contained in 10 CFR 50 provided that:

1. **The requested exemption is authorized by law.** 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. **The requested exemption does not present an undue risk to the public health and safety.** The proposed P/T limits rely in part on the requested exemption. The proposed P/T limits have been developed using the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. Margins that exist in the ASME Section XI, Appendix G P/T limit determination process are unaffected by this request.

   Use of the K_{IC} curve in determining the lower bound fracture toughness in the development of P/T operating limit curves is more realistic than the assumption under the use of the K_{IA} curve. The K_{IC} curve models the slow heatup and cooldown process of a reactor coolant system, with the fastest rate allowed being 100 °F per hour. The rate of change of pressure and temperature is often constant in this case. Both the heatup and cooldown and pressure testing are essentially static processes. During development of Code Case N-640 and the accompanying Appendix G code change, the ASME Section XI, Working Group on Operating Plant Criteria (WGOPC), performed assessments of margins inherent to K_{IA} using realistic heatup and cooldown curves. These assessments led to the conclusion that utilization of the K_{IA} curve was excessively conservative and the K_{IC} curve provided adequate margin for protection from brittle fracture.

The K_{IA} curve was codified in 1974. The initial K_{IA} conservatism was necessary due to limited experience and knowledge of the fracture toughness of reactor pressure vessel materials over time. The conservatism also provided margin thought to be necessary to cover uncertainties and a number of postulated but unquantified effects. Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that had been subject to the effects of neutron embrittlement in both an operating and test environment. The K_{IA} curve was based on 125 data points. The K_{IC} curve is based on more than 1500 data points. The additional data has significantly reduced the uncertainties associated with embrittlement effects and reduced other uncertainties. The added data ensures that the K_{IC} curve adequately and statistically bounds the data. The new information indicates the lower bound on fracture toughness provided by the K_{IA} curve is extremely conservative and is well beyond the margin of safety from potential reactor pressure vessel failure.

3. **The requested exemption will not endanger the common defense and security.** This request does not modify any physical plant architectural features, surveillance or alarm features. Therefore, 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 50.60and 10 CFR 50 Appendix G.** 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 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 is not necessary to achieve the underlying purpose of the rule;

   (a)(2)(iii) - Compliance with the regulation would result in undue hardship or other cost that are significant;

   (a)(2)(v) - The exemption would provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulation.

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

ASME Section XI, Appendix G provides the methodology for determining allowable P/T limits and is approved for that purpose by 10 CFR 50, Appendix G. Application of this methodology satisfies the underlying requirement for: 1) The RCS pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and 2) P/T limits provide adequate margin in consideration of uncertainties in determining the effects or irradiation on material properties.

ASME Section XI, Appendix G methodology was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of irradiation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements via application of ASME Code Case N-640 while maintaining the underlying purpose of the ASME code and the NRC regulations to ensure an acceptable margin of safety.

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

The RCS P/T operating window is defined by the P/T limit curve developed in accordance with the ASME Section XI, Appendix G methodology and the minimum P/T curve for pump operation. Continued operation of Catawba with these P/T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the operating window that results from these operating P/T limits. This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-640 in the development of the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety.

10 CFR 50.12(a)(2)(v):

The exemption provides only temporary relief from the applicable regulation and Catawba Nuclear Station has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry. However, to maintain sufficient operating margin to the end of the proposed Catawba Nuclear Station Units 1 and 2 pressure-temperature limits, we require an exemption to use ASME Code Case N-640.

ASME Code Case N-640, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 and 10 CFR 50 Appendix G would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the fracture toughness lower bound used by ASME Section XI, Appendix G in the determination of RCS P/T limits. This proposed alternative is acceptable because it reduces the excess conservatism in the current Appendix G. The safety margin that exists with the revised methodology is still very large. Restrictions on allowable operating conditions and equipment operability requirements are established to ensure RCS pressure and temperature is within the heatup and cooldown rate dependent P/T limits specified in TS 3.4.3. Therefore, this exemption does not present an undue risk to the public health and safety.

BC 98-379
ISI 94-004
Dec. '98

CASE
N-640

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: February 26, 1999
See Numeric Index for expiration and any reaffirmation dates.

Case N-640
Alternative Reference Fracture Toughness for Development of P-T Limit Curves
Section XI, Division 1

Inquiry: May the reference fracture toughness curve K_{IC}, as found in Appendix A of Section XI, be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves?

Reply: It is the opinion of the Committee that the reference fracture toughness K_{IC} of Fig. A-4200-1 of Appendix A may be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves. When this Case is employed LTOP Systems shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the the P-T Limit Curves.

TECHNICAL BASIS FOR REVISED P-T LIMIT CURVE METHODOLOGY

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ABSTRACT

The heatup and cooldown processes for an operating nuclear plant are controlled by pressure-temperature (P-T) limit curves, which are developed based on fracture mechanics analysis. These limits are developed according to Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI, and incorporate a number of safety margins. A key safety margin is the lower bound fracture toughness curve, or K_{IA} (equivalent to K_{IR}). Based on the work described in this technical basis paper, Section XI recently approved Code Case N-640 permitting the use of the lower bound static fracture toughness curve, K_{IC}, for calculating operating P-T limit curves. The same change appears in Appendix G of Section XI in the 1999 Addenda.

There are two lower bound fracture toughness curves available in Section XI. K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. Code Case N-640 changes the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By increasing the operating window relative to pump seal requirements, the chances of damaging pump seals and initiating a small loss of coolant accident (LOCA), a potential pressurized thermal shock (PTS) initiator, are reduced. Also, excessive shielding (e.g , dummy fuel assemblies on the corners of the core) to provide an acceptable operating window for current requirements can result in higher fuel peaking temperatures (due to changes in core power density) and less margin to fuel damage during an accident condition In addition, artificially high leak test temperatures (i.e , above 212°F) in boiling water reactors (BWRs) can be eliminated, which further increases plant safety.

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI P-T limit curve methodology. The safety margin that exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk. This paper describes the technical basis for the revised P-T limit curve methodology and presents the results of sample problems for both pressurized water reactors (PWRs) and BWRs.

INTRODUCTION

The startup and shutdown process, as well as pressure testing, for an operating nuclear plant is controlled by P-T limit curves, developed based on fracture mechanics analysis The methodology is defined in Appendix G of Section XI of the ASME Code, and incorporates four specific safety margins:

1. Large postulated flaw, ¼ thickness (¼-T) in vessel shell.
2. Safety factor = 2 on pressure stress for startup and shutdown.
3. Lower bound fracture toughness (KIA)
4. Upper bound adjusted reference temperature (RTNDT)

Although the above four safety margins were originally included in the methodology used to develop P-T limit curves and hydrotest temperatures, some sources of stress were not considered in the original methodology. These stresses include weld residual stresses and stresses due to clad-base metal differential thermal expansion. Furthermore, the original methodology assumed that the maximum value of the computed stress intensity factor occurred at the deepest point of the flaw.

Therefore, these elements required consideration to assess their effects on safety margins and justify the use of K_{IC}.

There are a number of reasons why the limiting toughness in the Appendix G P-T limits was changed from K_{IA} to K_{IC}. Each of these is described in the following paragraphs.

USE OF K_{IC} IS MORE TECHNICALLY CORRECT

The heatup and cooldown processes for nuclear plants are very slow, with the fastest rate allowed typically being 100°F per hour. For this rate of temperature change, the rate of change of pressure and temperature is often constant, so the resulting stresses are essentially constant. Therefore, both the heatup and cooldown processes, as well as pressure test conditions that have little or no thermal stress, are essentially static processes. In fact, with regard to fracture toughness, all operating transients (levels A, B, C and D) correspond to static loading conditions.

The only time when dynamic loading can occur and where the dynamic/arrest fracture toughness, K_{IA}, should be used for the reactor pressure vessel (RPV), is when a crack is propagating This situation may be postulated during a PTS transient event, but is not a credible scenario during the heatup or cooldown processes. Therefore, use of the static lower bound fracture toughness, K_{IC}, is more technically correct for development of P-T limit curves

USE OF HISTORICALLY LARGE MARGIN IS NO LONGER NECESSARY

In 1974, when the Appendix G methodology was first approved for use and implemented into the ASME Code, K_{IA} (K_{IR} in the terminology of the time) was used to provide additional margin thought to be necessary to cover uncertainties (e g., flawsize fracture toughness), as well as a number of postulated (but unquantified) effects (e.g , local brittle zones). Almost 25 years later, significantly more information is known about these uncertainties and effects.

FLAW SIZE

With regard to flaw indications in RPVs, there have been no indications found at the inside surface of any operating reactor in the core region, which exceed the acceptance standards of ASME Code Section XI, in the entire 28-year history of Section XI. This is a particularly impressive conclusion considering that core region inspections have been required to concentrate on the inner surface and near inner surface region since the implementation of U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.150 in 1983. Flaws have been found, but all have been qualified as buried, or embedded.

There are a number of reasons why no surface flaws exist, and these are related to the fabrication and inspection practices for vessels For the base metal and full penetration welds, a full volumetric examination and surface exam is required before cladding is applied, and these exams are repeated after cladding deposition.

Further confirmation of the lack of any surface indications has recently been obtained by the destructive examination of portions of several commercial RPVs, such as the Midland vessel and the PVRUF vessel

FRACTURE TOUGHNESS

Since the original formulation of the K_{IA} and K_{IC} fracture toughness curves in 1972, the fracture toughness database has increased by more than an order of magnitude, and both K_{IA} and K_{IC} remain lower bound curves. This is shown in Figure 1 for K_{IC} [1], compared to Figure 2, which is the original database [2]. In addition, the temperature range over which the data have been obtained has been extended to both higher and lower temperatures than the original database.

It can be seen from Figure 1 that there are a few data points which fall just below the K_{IC} curve. Consideration of these points, as well as the many (over 1,500) points above the curve, leads to the conclusion that the K_{IC} curve is a lower bound for a large percentage of the data. An example set of carefully screened data in the extreme range of lower temperatures is shown in Figure 3 [3]. The Data in Figures 2 and 3 satisfy K_{IC} validity limits in ASTM E399 standard for cleavage fracture toughness, whereas the K_{JC} data in Figure 1 include significant ductile tearing in the higher temperature data points.

LOCAL BRITTLE ZONES

Another argument for the use of K_{IA} in the original version of Appendix G was based upon the concern that there could be a small, local brittle zone in the weld or heat-affected-zone of the base material that could pop-in and produce a dynamically moving cleavage crack. Therefore, the toughness property used to assess the moving crack should be related to dynamic or crack arrest conditions, especially for a ferritic RPV steel showing distinct temperature and loading-rate (strain-rate) dependence. The dynamic crack should arrest at a quarter-thickness depth (¼-T), and any re-initiation should consider the effects of a minimum toughness associated with dynamic loading. This argument provided a rationale for assuming a ¼-T postulated flaw size and a lower bound fracture toughness curve considering dynamic and crack arrest loading. The K_{IR} curve in Appendix G of Section III, and the equivalent K_{IA} curve in Appendices A and G of Section XI provide this lower bound curve for high-rate loading (above any realistic rates in RPVs during any accident condition) and crack arrest conditions. This argument, of course, relies upon the existence of a local brittle zone.

After over 30 years of research on RPV steels fabricated under tight controls, micro-cleavage pop-in has not been found to be significant. This means that researchers have not produced catastrophic failure of a vessel, component, or even a fracture toughness test specimen in the transition temperature regime. The quality of quenched, tempered, and stress-relieved nuclear RPV steels, which typically have a lower bainitic microstructure, is such that there may not be any local brittle zones that can be identified. Testing of some specimens at Oak Ridge National Laboratory (ORNL) [4] has shown some evidence of early pop-ins for some simulated production weld metals. However, the level of fracture toughness for these possible early initiations is within the data scatter for other ASTM-defined fracture toughness values (K_{IC} and/or K_{JC}) Therefore, there is excess conservatism associated with this postulated condition and the use of the lower bound K_{IA} curve to assess fracture initiation This conservatism leads to unneeded margin that reduces overall plant safety

OVERALL PLANT SAFETY IS IMPROVED

The primary reason for developing Code Case N-640 was to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument exists that the proposed change will increase plant safety and reduce personnel exposure for both PWRs and BWRs.

Impact on PWRs:

By increasing the operating window relative to reactor coolant system (RCS) pump seal requirements, as shown schematically in Figure 4, the chances of damaging the seals from insufficient cooling water pressure and thereby initiating a small LOCA (a potential PTS initiator) are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements results in higher fuel peaking and less margin to fuel damage during an accident condition.

The use of K_{IC} also reduces the need for lock-out of the high pressure safety injection (HPSI) systems, which is usually done during low temperature operation when the system is water solid and residual heat removal (RHR) systems are in place to avoid a low temperature over pressure protection (LTOP) event and associated high pressure spike. Eliminating HPSI lock-out improves personnel and plant safety, and reduces the potential for a radioactive release. Finally, challenges to the plant low temperature overpressure protection (LTOP) system and potential problems with reseating the valves are also reduced.

Impact on BWRs:

The primary impact on the BWR is a reduction in the pressure test temperature. BWRs use recirculation pump heat to reach the required pressure test temperatures Several BWR plants are required to perform the pressure test at temperatures over 212°F under the current Appendix G criteria. Such high test temperatures result in several concerns: (i) pump cavitation and seal degradation may occur, (ii) primary containment isolation is required and ECCS/safety systems have to be operational at temperatures in excess of 212°F, and (iii) leak detection is difficult and more dangerous than at lower temperatures since the resulting leakage is steam and therefore poses safety hazards of burns and exposure to personnel The reduced test temperature eliminates these safety issues without reducing overall fracture margin.

REACTOR VESSEL FRACTURE MARGINS

To demonstrate the effect of the proposed Code Case, a series of P-T limit curves were produced for typical PWR and BWR plants. These curves were produced using identical input information, using both K_{IA} and the proposed new approach, K_{IC}. Since the limiting conditions for the PWR (cool-down) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs

It has long been known that the P-T limit curve methodology is very conservative [5,6] Changing the reference fracture toughness to K_{IC} maintains a very high margin, as illustrated in Figure 5 for a typical PWR. Similar results are shown for a BWR hydrotest in Figure 6. These figures each show a series of P-T curves using different assumptions concerning flaw size, safety margin, and fracture toughness. Reasonable assumptions for flaw size, weld residual stress, and clad residual stress (Reference Cases 1-3) yield allowable pressures significantly higher than previous Code methodology (ASME Appendix G with K_{IA} = (P-T Curve Case #1).

The results shown in Figures 5 and 6 were obtained from sample problems that were solved by several members of the Section XI Working Group on Operating Plant Criteria. The sample problems were developed to allow comparison calculations to be carried out to ensure the accuracy of the supporting calculations for the proposed change from K_{IA} to K_{IC}. The PWR problems were developed during several meetings at the NRC involving NRC staff and several members of the Section XI Working Group on Operating Plant Criteria [7]. Two major issues regarding the reference surface-crack depth (1-inch) and the use of a mean K_{IC} curve [3] were discussed in these meetings. The justification for selecting a 1-inch reference surface-crack depth is the reasonable assurance that current state-of-art NDE methods can reliably detect 1-inch deep surface crack indications in an RPV. The mean K_{IC} curve [3] was used for P-T curve deterministic analysis to provide a realistic best estimate for comparison with the ASME Code bounding toughness approach.

Later, a additional PWR problem was developed based on input from Working Group members, and similar problems were developed for application to BWRs. The sample problems required development of an operating P-T cooldown curve (for a PWR) or the pressure test curve (for a BWR) for irradiated RPV material.

The sample problems involved a tightly specified reference case, with three variations (four for the PWR), and then two P-T limit curve calculations whose input was also tightly specified, one using K_{IA} and the second using K_{IC}. The goal of the problems was to determine the margin on pressure which exists using the K_{IA} approach, and the margin which exists with the K_{IC} approach. The variations in the problems were intended to show the individual effects on margin from weld residual stresses, clad residual stresses, a smaller flaw (i.e., 1 inch), and K_{IC}. Each of the cases is briefly described below:

Reference Case #1:

A best-estimate P-T cooldown curve was determined for a typical PWR (a hydrotest curve was determined for a BWR) over the entire temperature range of operation, starting at 70°F. This problem was meant to be a best estimate curve with no specific safety factors, and best estimate values for each of the variables. Only pressure and thermal stresses were considered. The crack driving force, K_I, was computed along the entire crack-front and compared against the mean K_{IC} curve [3].

Reference Case #2:

This case is similar to Reference Case #1, but the weld residual stresses were included for a longitudinal weld in the RPV. The crack driving force, K_I, was computed along the entire crack-front and compared against the mean K_{IC} curve [3]

Reference Case #3:

This case is similar to Case #2, but clad residual stresses were included (see Figure 7), and the ASME Code K_{IC} curve was used. The crack driving force, K_I, was computed along the entire crack-front and compared against the mean K_{IC} curve [3].

Reference Case #4:

This case is similar to Case #3, except that the ASME Code K_{IC} curve was used. This case was run for a typical PWR only.

In addition to the four reference cases described above, two P-T Curve Cases were considered, as follows:

P-T Curve Case #1:

This case considered a classic P-T curve calculation done according to the existing rules in ASME Code, Section XI, Appendix G, using the KIA curve and the ASME Code specified safety factors. The crack driving force, KI, was computed at the deepest point of a ¼-T depth surface crack.

P-T Curve Case #2:

This case was the same as P-T Curve Case #1, except that the ASME Code K_{IC} curve was used.

Other parameters, such as leak test and bolt-up temperatures, were not calculated for the sample problems. The input variables for the sample problems are shown in Tables 1 through 3.

For each case, a pressure ratio was computed with respect to P-T Curve Case #1 (which represents the previously existing Section XI methodology) to determine the margins (pressure ratios) that are included in the curves. Typical results are shown in Table 4 and Figure 8 for the PWR problem Similar results were achieved for the BWR problem, as shown in Table 5 and Figure 9.

Comparing the reference or best estimate curve with the two P-T curves calculated using code requirements in each figure, it is seen that there is significant margin on the allowable pressure for both the K_{IA} or K_{IC} cases. The same is true for Reference Case #4 for the PWR problem where the ASME K_{IC} curve (rather than a mean K_{IC} curve) was used.

For PWRs, another important contribution to the margin, is the LTOP system, which is operational in the low temperature range. The margins increase significantly for higher temperatures, as seen in Figure 5.

CONCLUSION

Technology and data developed over the last 25 years has provided a strong basis for revising the ASME Section XI P-T limit curve methodology. Based on the pressure ratios shown in Tables 4 and 5 the safety margin that exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk. This was demonstrated via sample problems for both PWRs and BWRs that considered weld residual stress, clad residual stress, a smaller flaw size, and K_{IC}.

Changing the methodology results in an overall increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease, and the personnel safety issues associated with artificially high test temperatures will be eliminated.

ACKNOWLEDGEMENT

The authors would like to acknowledge the members of the ASME Code Section XI Working Group on Operating Plant Criteria for their dedicated efforts in working the sample problems and developing Code Case N-640.

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Table 1
Input Parameters for Reference Cases

Parameter	PWR Value	BWR Value
Thickness	9.0 inches	6.0 inches
Inside Radius	90 inches	125 inches
Clad Thickness	0.25 inch	0.25 inch
Mean K_{IC} (Ref. Cases #1, 2, and 3 only)	36 36 + 51.59 exp [0.0115 (T-RT_{NDT})]	36.36 + 51.59 exp [0.0115 (T-RT_{NDT})]
Flaw Size	1" deep, 6" long	1" deep, 6" long
Event	Cooldown: 100°F/hr from 550°F to 200°F, 20°F/hr from 200°F to 70°F	Pressure Test
RT_{NDT}	236°F (inside surface)	168°F (flaw tip)
Heat Transfer Coefficient	1,000 BTU/hr-ft^2-°F	N/A
Weld Residual Stress (Ref. Case #2, 3, and 4 only)	see Table 3	see Table 3
Clad Residual Stress (Ref. Case #3 and 4 only)	see Figure 7	see Figure 7

Table 2
Input Parameters for P-T Curve Cases

Parameter	PWR Value	BWR Value
Thickness	9 0 inches	6.0 inches
Inside Radius	90 inches	125 inches
Clad Thickness	0.25 inch	0.25 inch
Flaw Size	1" deep, 6" long	1" deep, 6" long
Event	Cooldown: 100°F/hr from 550°F to 200°F, 20°F/hr from 200°F to 70°F	Pressure Test
RT_{NDT}	236°F (inside surface)	168°F (inside surface)
Heat Transfer Coefficient	1,000 BTU/hr-ft^2-°F	N/A

Table 3
Weld Residual Stress Distribution for Reference Cases #2, #3, and #4

Depth (a/T)	Stress (ksi)
0.000	6.50
0.045	5.47
0.067	4.87
0.101	3.95
0.134	2.88
0.168	1.64
0.226	-0.79
0.285	-3.06
0.343	-4.35
0.402	-4.31
0.460	-3.51
0.510	-2.57
0.572	-1.70
0.619	-1.05
0.667	-0.46
0.739	0.35
0.786	0.87
0 834	1.41
0.881	1.96
0.929	2.55
0.976	3.20
1.000	3.54

Table 4
Summary of Allowable Pressures for a 20°F/hr Cooldown at 70°F and RT_{NDT} 236°F (Typical PWR Plant)

Type of Evaluation	Allowable Pressure(1) (psi)	Pressure Ratio(2)
P-T Curve Case #1: Appendix G with 1/4-T flaw and K_{IA} Limit.	420	1.00
P-T Curve Case #2: Appendix G with 1/4-T flaw and K_{IC} Limit.	530	1.26
Reference Case #1: 1-inch flaw for pressure and thermal loading only, and mean K_{IC} curve [3].	2,305	5.48
Reference Case #2: 1-inch flaw for pressure, thermal, residual loads, and mean K_{IC} curve [3].	1,845	4.38
Reference Case #3: 1-inch flaw for pressure, thermal, residual, and cladding loads, and mean K_{IC} curve [3].	1,520	3.61
Reference Case #4: 1-inch flaw for pressure, thermal, residual, and cladding loads, and ASME K_{IC} curve.	800	1.90

Note: 1. Comparable values of allowable pressure were calculated by various members of the ASME Section XI Working Group on Operating Plant Criteria.
2. The pressure ratio equals the allowable pressure of the case in question divided by the allowable pressure of the Base Case (P-T Curve Case #1). This demonstrates the margins inherent to previous Section XI, Appendix G methods.

Table 5
Summary of Allowable Pressures for Primary Hydrotest at 70°F and RT_{NDT} of 168°F (Typical BWR Plant)

Type of Evaluation	Allowable Pressure(1) (psi)	Pressure Ratio(2)
P-T Curve Case #1: Appendix G with 1/4-T flaw and K_{IA} Limit.	530	1.00
P-T Curve Case #2: Appendix G with 1/4-T flaw and K_{IC} Limit.	648	1.22
Reference Case #1: 1-inch flaw for pressure and thermal loading only, and mean K_{IC} curve [3].	1380	2.60
Reference Case #2: 1-inch flaw for pressure, thermal, residual loads, and mean K_{IC} curve [3].	1220	2.30
Reference Case #3: 1-inch flaw for pressure, thermal, residual, and cladding loads, and mean K_{IC} curve [3]	825	1.55

Note: 1. Comparable values of allowable pressure were calculated by various members of the ASME Section XI Working Group on Operating Plant Criteria.
2. The pressure ratio equals the allowable pressure of the case in question divided by the allowable pressure of the Base Case (P-T Curve Case #1). This demonstrates the margins inherent to previous Section XI, Appendix G methods.

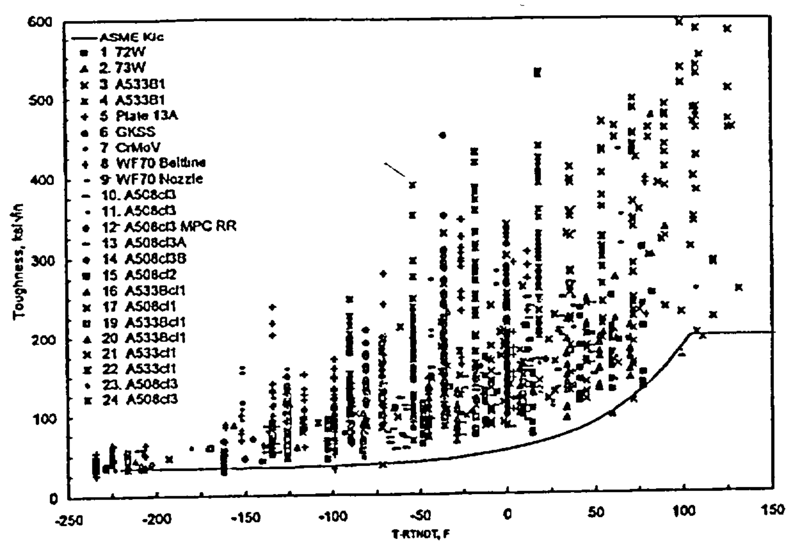

Figure 1
Static Fracture Toughness Data (K_{JC}) Now Available, Compared to K_{IC} [1]

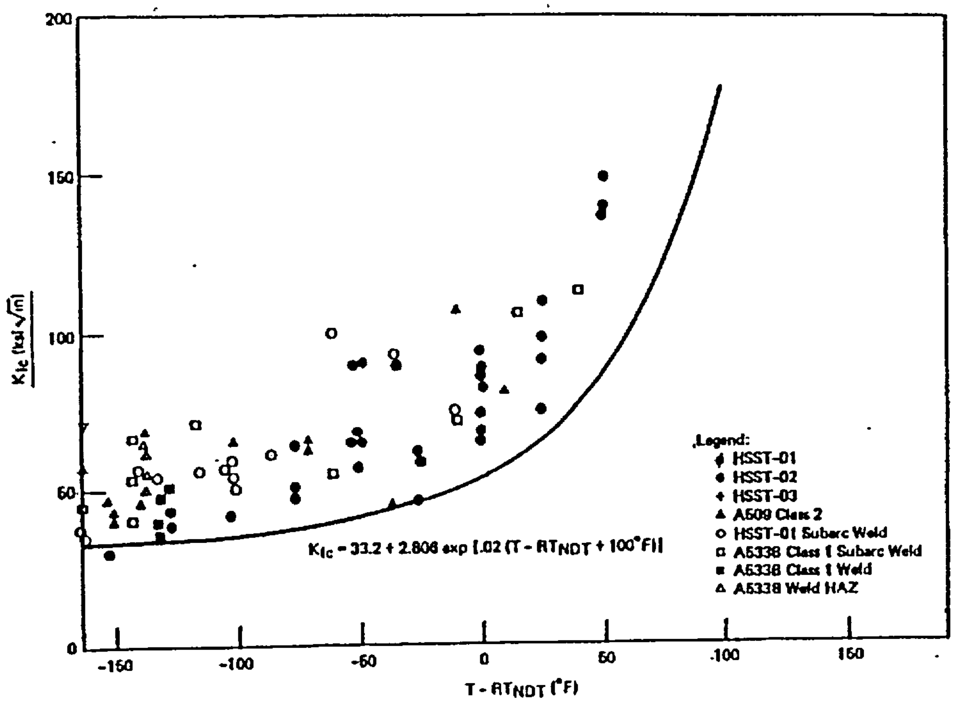

Figure 2
Original K_{IC} Reference Toughness Curve, with Supporting Data [2]

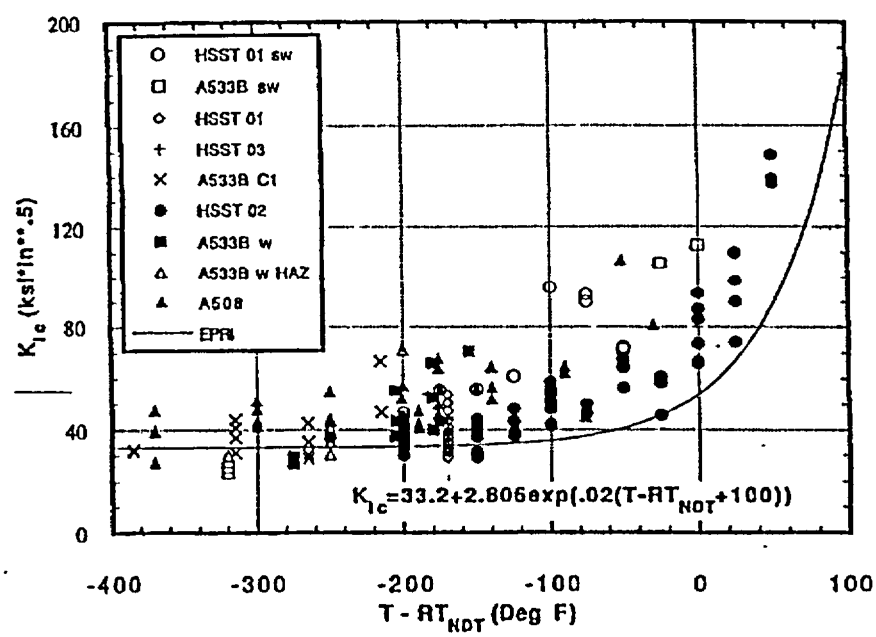

Figure 3
K_{IC} Reference Toughness Curve with Screened Data in the Lower Temperature Range [3]

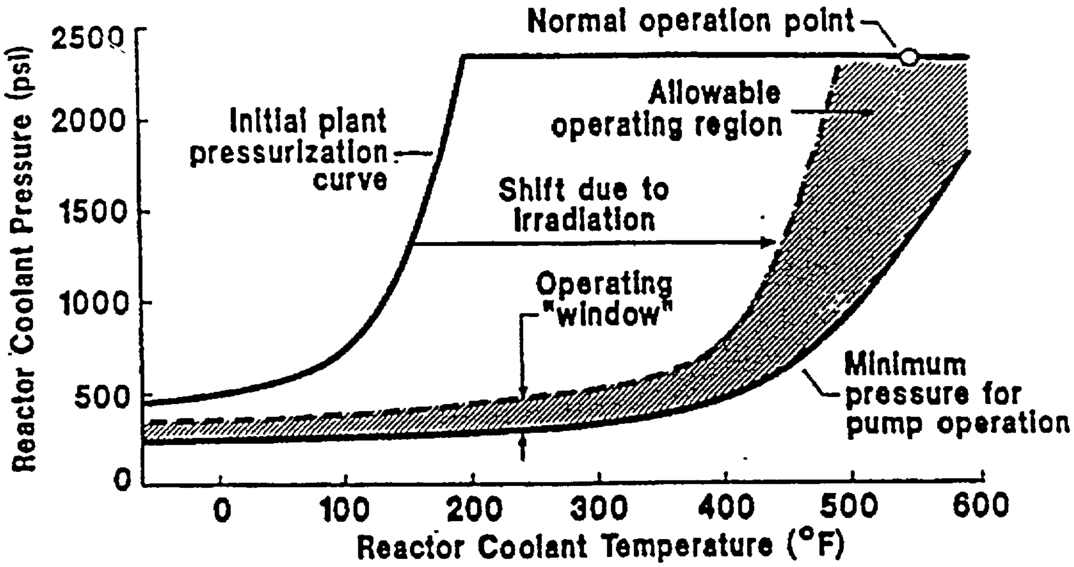

Figure 4
Operating Window From P-T Limit Curves [4]

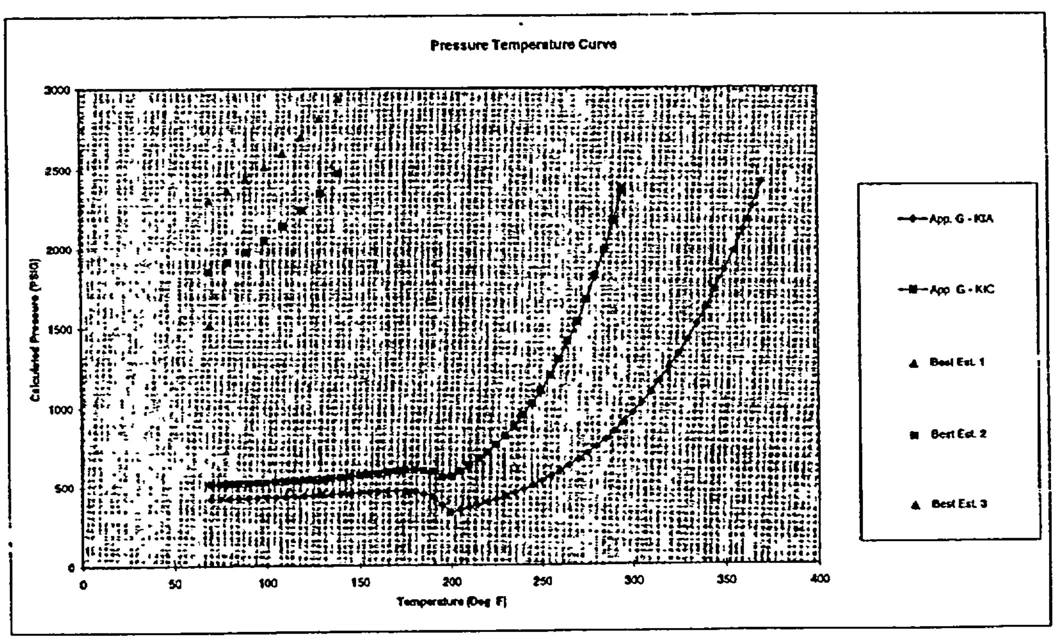

Figure 5
P-T Limit Curves Illustrating Deterministic Safety Factors for a PWR Reactor Vessel

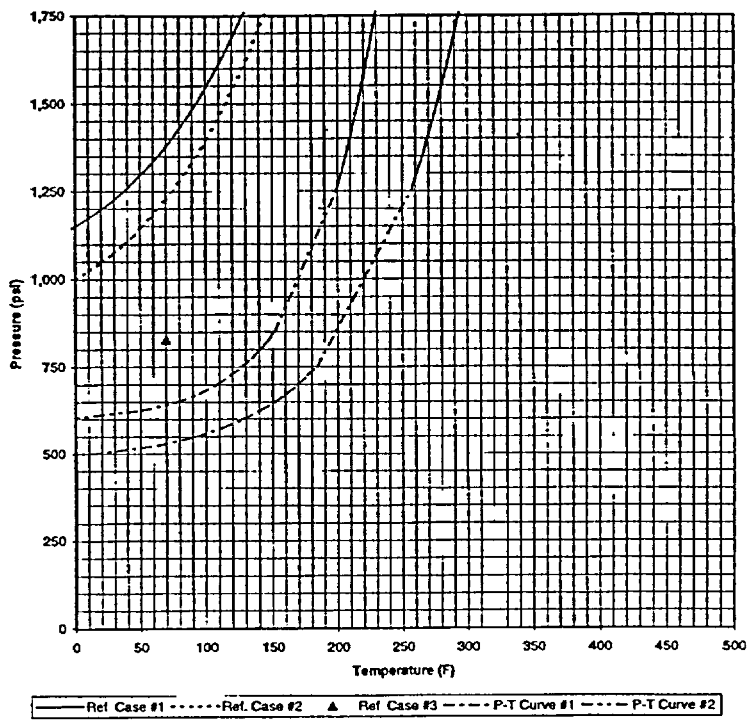

Figure 6
P-T Limit Curves Illustrating Deterministic Safety Factors for a BWR Reactor Vessel

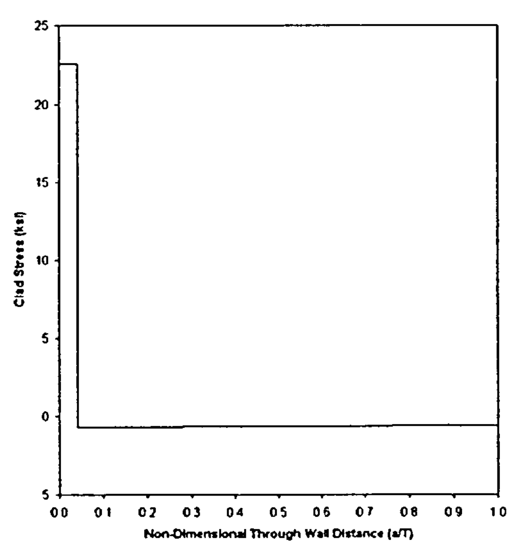

Figure 7
Clad Residual Stress Distribution for Reference Cases #3 and #4

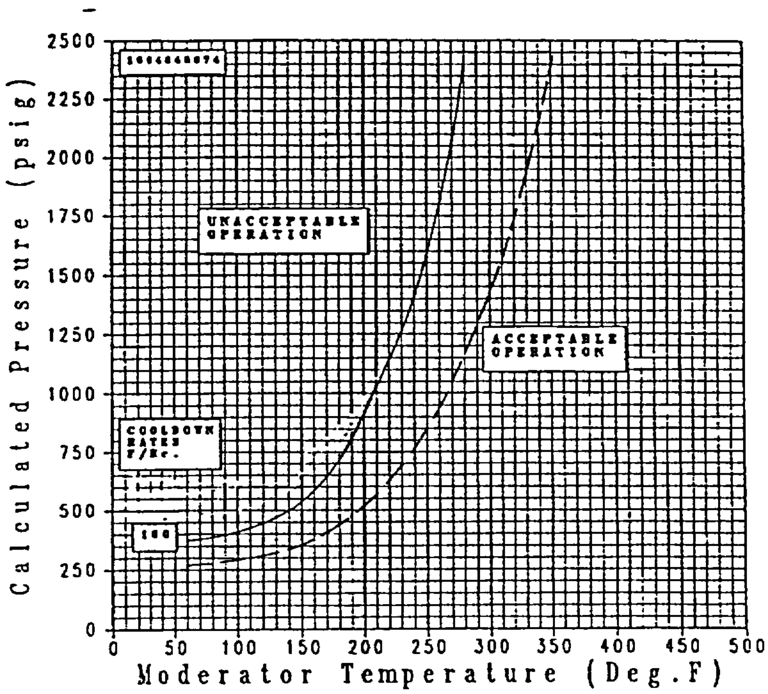

Figure 8
Comparison of Cool-Down Curves for a PWR for the Existing and Proposed Methods
(Dashed Curve = K_{IA} and Solid Curve K_{IC})

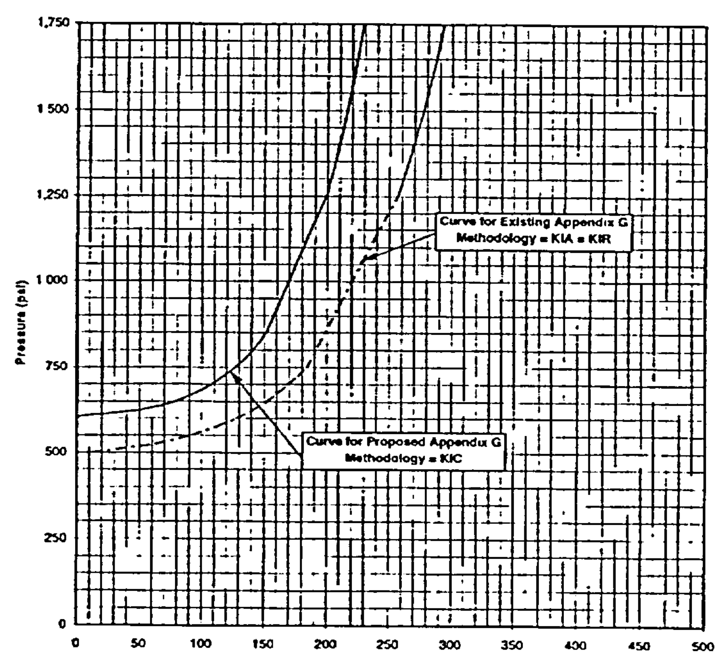

Figure 9
Comparison of Hydrotest P-T Curves for a BWR for the Existing and Proposed Methods

ATTACHMENT 6

Justification for ASME Code Case N-641 Exemption Request

Justification for ASME Code Case N-641 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection (LTOP) System Requirements, Section XI, Division 1," in lieu of the methods specified in 10 CFR 50, Appendix G.

Compliance with 10 CFR 50.12 Requirements:

The requested exemption to allow the use of ASME Code Case N-641 to determine the LTOP system enable temperature meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states the Commission may grant an exemption from the requirements contained in 10 CFR 50 provided that:

1. **The requested exemption is authorized by law.** 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. **The requested exemption does not present an undue risk to the public health and safety.** 10 CFR 50, Appendix G, requires, in part, that Article G-2215 of ASME XI, Appendix G, be used to determine the effective coolant temperature range of the LTOP system. Article G-2215 states that for plants that have LTOP systems, the system shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} +50 °F, whichever, is greater. This temperature is based on an axially oriented flaw.

   The revised LTOP enable temperature being proposed for Catawba Nuclear Station (CNS) Units 1 & 2 was developed using the methodology provided in Code Case N-641. Use of the Code Case N-641 methodology in the determination of the LTOP enable temperature is more technically correct than the generic value included in earlier versions of ASME Section XI and reduces inconsistencies in the margin of safety between reactor vessel geometries.

   The basis for the enable temperature in ASME Code Case N-641 provides bounding reactor vessel low temperature integrity protection during LTOP design basis transients. The LTOP PORV setpoint utilizes 100% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Section XI, Division 1, as a design limit. This approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

3. **The requested exemption will not endanger the common defense and security.** This request does not modify any physical plant architectural features, surveillance or alarm features. Therefore, 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 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 is not necessary to achieve the underlying purpose of the rule;

   (a)(2)(iii) - Compliance with the regulation would result in undue hardship or other cost that are significant;

   (a)(2)(v) - The exemption would provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulation.

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

The underlying purpose of 10 CFR 50, Appendix G and ASME Section XI, Appendix G, is to satisfy the requirement that: (1) the reactor coolant system (RCS) pressure boundary be operated in a regime having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (2) Pressure-Temperature operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

Application of Code Case N-641 to determine the LTOP T_{enable} provides appropriate procedures to determine the limiting temperature below which, protection is required against overpressure conditions. Sufficient margin remains to assure the CNS reactor vessels behave in a non-brittle manner.

Implementation of an LTOP enable temperature without the additional margin associated with ASME Code Case N-641 would unnecessarily restrict the pressure-temperature operating window. The LTOP enable temperature established in accordance with ASME Code Case N-641 will minimize the time spent in LTOP operation and reduce the risk associated with undesired actuation of LTOP. Therefore, use of Code Case N-641, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

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

The reactor coolant system pressure-temperature operating window is defined by the pressure-temperature operating and test curves developed in accordance with the ASME Section XI, Appendix G procedures. Operation with these pressure-temperature curves without the relief provided by ASME Code Case N-641 would unnecessarily restrict the pressure-temperature operating window for CNS Units 1 and 2 and increase the amount of time the units are operated in LTOP mode. The proposed LTOP guidelines will increase the operating window by lowering the temperature regime in which LTOP is operable.

The current methodology provides a restrictive enable temperature which constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-641. Implementation of the proposed enable temperature as allowed by ASME Code Case N-641 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

10 CFR 50.12(a)(2)(v)

The exemption provides only temporary relief from the applicable regulation and CNS has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time that the NRC generically approves ASME Code Case N-641 for use by the nuclear industry. However, to maintain sufficient operating margin to the end of the proposed CNS Units 1 and 2 pressure-temperature and LTOP limits, we require an exemption to use ASME Code Case N-641.

ASME Code Case N-641, Conclusion for Exemption Acceptability:

Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-641 presents a benefit in safety to the public in that the units are operated for less time in LTOP mode. Implementation of the ASME Code Case N-641 analysis methodology for setting the LTOP enable temperature, by using plant specific determination, ensures that the bounding protection of ASME Section XI, Appendix G limits is provided.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, reactor coolant system pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in Technical Specifications. Therefore, this exemption does not present an undue risk to public health and safety.

Approval Date: January 17, 2000
See Numeric Index for expiration and any reaffirmation dates.

Case N-641
Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements
Section XI, Division 1

Inquiry: What alternatives to Appendix G-2215 may be used for determination of pressure-temperature relationships and low temperature overpressure protection system effective temperatures and allowable pressures?

Reply: It is the opinion of the Committee that, as an alternative to Appendix G-2215, the following may be used.

-1000 INTRODUCTION

-1100 Scope

This Case presents alternative procedures for calculating pressure-temperature relationships and low temperature overpressure protection (LTOP) system effective temperatures and allowable pressures. These procedures take into account alternative fracture toughness properties, circumferential and axial reference flaws, and plant-specific LTOP effective temperature calculations.

-2215 Allowable Pressure

-2215.1 Pressure-Temperature Relationship. The equations below provide the basis for determination of the allowable pressure at any temperature at the depth of the postulated defect during Service Conditions for which Level A and Level B Service Limits are specified. In addition to the conservatism of these assumptions, it is recommended that a factor of 2 be applied to the calculated K_I values produced by primary stresses. In shell and head regions remote from discontinuities, the only significant loadings are: (1) general primary membrane stress due to pressure; and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Therefore, the requirement to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined is:

$$2K_{Im} + K_{It} < K_{Ic} \qquad (1)$$

throughout the life of the component at each temperature with K_{Im} from G-2214.1, K_{It} from G-2214.3, and K_{Ic} from Fig. G-2210-1.

The allowable pressure at any temperature shall be determined as follows.

(a) For the startup condition,

(1) consider postulated defects in accordance with G-2120;

(2) perform calculations for thermal stress intensity factors due to the specified range of heat-up rates from G-2214.3;

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding locations of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location. The allowable pressure-temperature relationship is the minimum pressure at any temperature determined from

(a) the calculated steady-state ($K_{It} = 0$) results for the $\frac{1}{4}$ thickness inside surface postulated defects using the equation:

$$P = \frac{K_{Ic}}{2M_m}\left({}^{t}/_{R_i}\right)$$

(b) the calculated results from all vessel beltline materials for the heatup stress intensity factors using the corresponding $\frac{1}{4}$ thickness outside-surface postulated defects and the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left({}^{t}/_{R_i}\right)$$

(b) For the cooldown condition;

(1) consider postulated defects in accordance with G-2120,

(2) perform calculations for thermal stress intensity factors due to the specified range of cooldown rates from G-2214.3;

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding location of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location using the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left({}^{t}/_{R_i}\right)$$

The allowable pressure-temperature relationship is the minimum pressure at any temperature, determined from all vessel beltline materials for the cooldown stress intensity factors using the corresponding $\frac{1}{4}$ thickness inside-surface postulated defects.

-2215.2 Low Temperature Overpressure Protection System. Plants having LTOP systems may use the following temperature and pressure conditions to provide protection against failure during reactor startup and shutdown operation due to low temperature overpressure events that have been classified Service Level A or B.

(a) LTOP System Effective Temperature. The LTOP system effective temperature T_e is the temperature at or above which the safety relief valves provide adequate protection against nonductile failure. LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (2) below. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) below.

(1) a coolant temperature[1] of 200°F;

(2) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is defined for inside axial surface flaws as RT_{NDT} + 40°F, and T_e is defined for inside circumferential surface flaws as RT_{NDT} − 85°F;

(3) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is calculated on a plant specific basis for the axial and circumferential reference flaws using the following equation:

$$T_e = RT_{NDT} + 50 \ln\left[\left(\left(F \cdot M_m \left(pR_i / t\right)\right) - 33.2\right) / 20.734\right]$$

where

F = 1.1, accumulation factor for safety relief valves

M_m = the value of M_m determined in accordance with G-2214.1

p = vessel design pressure, ksi

R_i = vessel inner radius, in.

t = vessel wall thickness, in.

(b) LTOP System Allowable Pressure. LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy Eq. (1) if K_{Ic} is used for determination of allowable pressure, or 110% of the pressure determined to satisfy Eq. (1) if K_{Ia} is used (as an alternative to K_{Ic}) for determination of allowable pressure.

[1] The coolant temperature is the reactor coolant inlet temperature.

[2] The vessel metal temperature is the temperature at a distance one-fourth of the vessel section thickness from the clad-base-metal interface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature, for weld or base metal in the beltline region, at a distance one-fourth of the vessel section thickness from the clad-base-metal interface as determined in accordance with Regulatory Guide 1.99, Rev. 2.

BASIS FOR PLANT-SPECIFIC LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM ENABLE TEMPERATURE

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ABSTRACT

As older pressurized water reactors (PWRs) with high copper welds approach the end of their operating licenses and make the transition to a license renewal period, the Low Temperature Overpressure Protection (LTOP) System effective or enable temperature (T_{enable}) must take into account:

- increased RT_{NDT} values due to radiation induced embrittlement of reactor vessel material;
- temperature differences between the coolant and the 1/4t reactor vessel location; and
- additional margins imposed or regulatory requirements, such as instrument uncertainty.

These factors will cause T_{enable} values to exceed 350°F for some plants. This will result in violation of the licensing and design basis for plants that require diverse means of low temperature overpressure relief using the residual heat removal (RHR) system relief valves. The RHR system is typically not designed for service above RCS temperatures of 350°F. Also, for plants which are required to operate shutdown cooling or decay heat removal systems at and below 300°F, T_{enable} values in this range increase complexity for the operators. As a means of maintaining acceptable margins of safety, satisfying the system licensing and design basis, and minimizing operational complexity, this paper demonstrates a method and provides the technical basis for determination of plant specific T_{enable} values for PWRs.

NOMENCLATURE

F = Safety margin on pressure for determination of T_{enable} temperature
K_{Ia} = Critical arrest stress intensity factor (ksi-in$^{1/2}$)
K_{Ic} = Critical initiation stress intensity factor (ksi-in$^{1/2}$)
K_{Im} = stress intensity factor due to membrane stress
K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
K_{It} = stress intensity factor due to thermal gradient (ksi-in$^{1/2}$)
LTOP = Low Pressure Overpressure Protection
M_m = Membrane stress correction factor
NRC = U.S. Nuclear Regulatory Commission
p = reactor vessel internal pressure (ksi)
R_i = vessel inner radius (in.)
RSB = NRC Reactor Systems Branch
RT_{NDT} = material adjusted reference temperature
t = vessel wall thickness (in.)
T_{enable} = Temperature at which LTOP systems must be effective or enabled

INTRODUCTION

NRC Branch Technical Position (BTP) RSB 5-2 was revised in 1988 to include guidance on determination of the enabling temperature for LTOP systems (USNRC [1]). In the years since, T_{enable} (or $T_{effective}$, as it has been designated in a recent ASME Section XI Code action) has become widely believed in the nuclear industry to be a fundamental material property, defined strictly by a margin from the material adjusted reference temperature (RT_{NDT}). Contrary to this, T_{enable} is a derived parameter based on several factors, including material fracture toughness, reactor pressure vessel dimensions, and the membrane stress intensity acting upon a postulated RPV surface flaw. Branch Technical Position RSB 5-2 specifies T_{enable} as the water temperature corresponding to a metal temperature of RT_{NDT} + 90°F.

The factor of safety on design pressure used in the determination of a T_{enable} temperature must not be confused with the 100% or 110% of allowable pressure at a given temperature permitted by ASME Section XI [2] for an LTOP pressure setpoint, depending on the reference fracture toughness used.

Gamble [4] published the basis for the definition of T_{enable} as RT_{NDT} + 50°F following the development of ASME Section XI Code Case N-514 [3]. This derivation of T_{enable} is based on determination of the temperature that would allow RCS pressure in a Westinghouse designed 4-loop RPV to reach 110% of the reactor vessel design pressure without initiation of the ASME Section XI maximum postulated flaw. Again, this factor of safety on design pressure for T_{enable} temperature determination is not related to the 100% or 110% of allowable pressure permitted for an LTOP pressure setpoint at a given temperature, which depends on the reference fracture toughness used.

The basis document for Code Case N-514 further demonstrates that T_{enable} is dependent upon the following parameters:

a) Irradiation embrittlement adjusted reference temperature (RT_{NDT}),
b) Vessel dimensions (inside radius and thickness exclusive of cladding);
c) Reference stress intensity factor (K_{Ic} or K_{Ia});
d) Pressure stress intensity factor; and
e) Safety margin provided on pressure stress intensity factor (1.0, 1.1, or 2.0).

This technical basis can be applied to calculate T_{enable} on a plant specific basis.

MAKING MARGINS OF SAFETY CONSISTENT

Another benefit that can be achieved by determination of plant specific T_{enable} values is the application of a consistent margin of safety to all PWRs for this parameter. The definitions for enable temperature currently in use, as specified in Code Case N-514 (which was incorporated into the 1993 Addenda of ASME Section XI Appendix G) or BTP RSB 5-2, result in inconsistent margins of safety for PWRs. This is because T_{enable} is dependent upon reactor vessel dimensions, and reactor vessels that are smaller than the reference case for Code Case N-514 (e.g., all Westinghouse designed 2-loop and 3-loop reactors) are penalized when using T_{enable} criteria established for protection of larger reactor pressure vessels.

With the publication of ASME Section XI Code Case N-588 [5], another parameter affecting T_{enable} was identified. Code Case N-588 allowed the reference flaw applied to circumferential welds to be oriented circumferentially rather than axially. The Code Case takes credit for the extremely low likelihood of a flaw being oriented in an axial manner within circumferential weldments. This results in another inconsistency in the T_{enable} margin of safety when this Code Case is applied, due to the effect of flaw orientation on allowable pressure. Because the currently defined values for T_{enable} are based on the stress intensity factor for an axially oriented reference flaw, the current definitions for T_{enable} are inadequate in plants where Code Case N-588 is applied.

The solution to the issue of inconsistent margin of safety is to develop and implement a method for determination of T_{enable} on a plant specific basis for any given pressurized water reactor vessel. This methodology will consider the factors identified above, most notably reactor vessel dimensions and postulated flaw orientation, and can be used to derive T_{enable} for each PWR vessel with a consistent and well defined margin of safety against brittle failure at low temperatures.

DESIGN BASIS FOR LTOP ENABLE TEMPERATURE

The design bases for T_{enable}, as defined in the basis document for Code Case N-514, were examined to document the assumptions and margins of safety implicit in this parameter. With this understanding, a plant specific approach to T_{enable} is defined using a consistent design basis, such that equivalent and consistent margins of safety are established for all PWR reactor vessels.

The basis document for Code Case N-514 defines the basis for the LTOP enabling temperature as:

> "The LTOP enabling temperature assessment involved determining the temperature that would allow the pressure to reach 110% of the design pressure, or typically about 2,750 psi for PWRs, without initiation of a postulated quarter-thickness depth flaw having RT_{NDT} at the tip of the flaw equal to 300°F. . . . The results are presented in Figure 3 and indicate that pressure greater than 110% of design pressure is achieved at a temperature equal to approximately RT_{NDT} + 50°F."

It should be noted that the statement "initiation of a postulated flaw," implies that the initiation fracture toughness, K_{Ic}, was utilized in this evaluation, in lieu of arrest fracture toughness, K_{Ia}. In fact, the Figure 3 that is referenced in the Code Case N-514 basis document notes that "Toughness = ASME K_{Ic}."

The Code Case N-514 basis document does not provide the specific underlying equations used to derive T_{enable}. However, using the information provided in the Code Case, it is possible to derive an explicit closed form solution for T_{enable}. This is provided below.

Derivation of Enabling Temperature – Code Case N-514

Based on ASME Section XI, Appendix G, G-2215 [2]:

$$K_{IR} > F \bullet K_{Im} + K_{It} \quad (1)$$

where:

K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
F = Safety margin on pressure for T_{enable} temperature determination
$K_{Im} = M_m \bullet (pR_i/t)$
K_{It} = 0, assuming isothermal conditions for LTOP
M_m = Membrane stress correction factor from ASME Section XI, Appendix G, Figure G-2214-1 (prior to 1996 Addenda)
p = internal pressure (ksi)
R_i = vessel inner radius (in.)
t = vessel wall thickness (in.)

In the basis document for Code Case N-514, the following parameters were selected:

$K_{Ic} = 33.2 + 20.734 \exp [0.02 (T - RT_{NDT})]$ is substituted for K_{IR} (the equation shown for K_{Ic} is taken from ASME Section XI, Appendix A, Article A-4200)
F = 1.1
p = 2.5 ksia
R_i = 86.9 inch
t = 8.9 inch
M_m = 2.87, Figure G-2214-1 (t = 8.9 inch, σ/σ_y = 0.5)

Substituting the above into Equation 1 and solving for T:

$$T = RT_{NDT} + 37.5°F \quad (2)$$

An additional margin was added to this result to round the additive term:

$$T_{enable} = RT_{NDT} + 50°F \quad (3)$$

Because the derivation of T_{enable} provided in the Code Case N-514 basis document was performed by somewhat graphical means, including an additional margin by rounding to RT_{NDT} + 50°F was reasonable to ensure that adequate safety margin was provided.

However, when T_{enable} is explicitly calculated using a closed form solution, this additional "windage" margin is not necessary; sufficient margin is derived from including the factor of 1.1 on pressure in the T_{enable} calculation. The margin on temperature provided by calculating the T_{enable} temperature as the temperature at which the allowable pressure is 110% of design pressure, can be illustrated by calculating the T_{enable} which would result at 100% of design pressure:

$$33.2 + 20.734 \exp [0.02 (T - RT_{NDT})] = [1.0 \bullet 2.87 \bullet 2.5 \bullet 86.9] / 8.9 \quad (4)$$

Which can be solved for T:

$$T = RT_{NDT} + 28.8°F \quad (5)$$

This results in a difference of 37.5 - 28.8, or 8.7°F. No additional margin on temperature is needed; the margin on pressure demonstrated in the Code Case N-514 basis document when the maximum pressure allowed by the LTOP system is 110% of the allowable pressure based on ASME Section XI Appendix G is already substantial, between 1.7 and 2.0. Since LTOP events are essentially isothermal, this margin on temperature is simply good engineering practice.

This case may also be evaluated using Westinghouse 2-loop reactor vessel dimensions (R_i = 66.16 inches, t = 6.5 inches) at a temperature of RT_{NDT} + 37.5°F, then solved for F (the safety margin on pressure). This results in a safety margin on pressure of 126% (utilizing the older Code stress intensity factors). This is significant in that it demonstrates the inconsistency of margin of safety based on a single generic enable temperature: at the same enable temperature, a large 4-loop RPV is protected against initiation of brittle failure to 2750 psig, while a 2-loop Westinghouse RPV is protected to 3143 psig. This represents a significant operating margin penalty on 2-loop reactors.

DERIVATION OF RELATION FOR PLANT SPECIFIC ENABLE TEMPERATURE

Using the methodology of Code Case N-514, it is possible to establish T_{enable} for any size RPV with a calculation using the methodology defined in the Code Case basis document. In addition, axial and circumferential flaw orientation will be considered in this evaluation by application of Code Case N-588.

Stress Intensity for a Postulated Surface Flaw

Based on ASME Section XI, Appendix G, G-2215 [2]:

$$K_{IR} > F \bullet K_{Im} + K_{It} \quad (6)$$

where:

K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
F = Safety margin on pressure for determination of T_{enable} temperature
$K_{Im} = M_m (pR_i/t)$
K_{It} = 0, assuming isothermal conditions for LTOP
p = internal pressure (ksi)
R_i = vessel inner radius (in.)
t = vessel wall thickness (in.)

The following parameters are selected to establish T_{enable}:

$K_{Ic} = 33.2 + 20.734 \exp [0.02 (T - RT_{NDT})]$ is substituted for K_{IR} (the equation shown for K_{IC} is taken from ASME Section XI Appendix A, Article A-4200)
F = 1.1 (basis for Code Case N-514 T_{enable} temperature)
p = vessel design pressure

Substituting and reducing:

$$33.2 + 20.734 \exp [0.02 (T - RT_{NDT})] = [1.1 \bullet M_m (pR_i/t)] \quad (7)$$

This leads to:

$$T = RT_{NDT} + 50 \ln [((1.1 \bullet M_m (pR_i/t)) - 33.2)/20.734] \quad (8)$$

Equation 8 establishes a relationship for determination of T_{enable} on a plant specific basis for any size RPV, and accounts for alternate postulated flaw orientations through the factor, M_m.

CALCULATION OF ENABLE TEMPERATURE FOR WESTINGHOUSE 2-LOOP REACTOR

Applying the plant-specific methodology above (along with the most recently available stress intensity factors from Code Case N-588 for axial and circumferential flaws) to a typical Westinghouse 2-Loop reactor, the LTOP system would be effective at coolant temperatures less than the greatest value of T_{enable} determined for 1) the most limiting axial flaw; 2) the most limiting circumferential flaw; and 3) 200°F

Inside Surface Axial Flaw

Solve Equation 8 for Westinghouse 2-Loop reactor dimensions assuming an inside surface (IS) axial flaw:

$M_m = 0.926\, t^{1/2}$ for IS axial flaw, $2 \le t^{1/2} \le 3.464$ (Code Case N-588)
p = vessel design pressure = 2.5 ksia
R_i = 66.16 inch
t = 6.5 inch

$$T = RT_{NDT} + 23.1°F \quad (9)$$

This result establishes the enable temperature based on a postulated axial flaw for a typical Westinghouse 2-Loop reactor vessel.

Inside Surface Circumferential Flaw

Solve Equation 8 for Westinghouse 2-Loop reactor dimensions assuming an inside surface circumferential flaw:

$M_m = 0.443\, t^{1/2}$ for IS circumferential flaw, $2 \le t^{1/2} \le 3.464$ (Code Case N-588)
p = vessel design pressure = 2.5 ksia
R_i = 66.16 inch
t = 6.5 inch

$$T = RT_{NDT} + 50 \ln [(31.6 - 33.2) / 20.734] \quad (10)$$

Equation 10 cannot be solved for T because the logarithm of a negative number would need to be taken. On a physical basis, this is because the minimum available initiation fracture toughness of reactor vessel steels at any temperature is always greater than the crack opening stress intensity on a circumferential reference flaw in a Westinghouse 2-loop reactor vessel at 110% of the design pressure, assuming isothermal conditions. Therefore, a circumferentially oriented reference flaw cannot initiate.

Based on this evaluation, Westinghouse 2-Loop reactor LTOP systems would be effective at coolant temperatures less than 200°F, or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 23°F for the most limiting of plates, forgings, and axial welds, whichever is greater. In this example, circumferential welds would never be controlling.

CIRCUMFERENTIAL AND AXIAL EVALUATION FOR OTHER VESSEL GEOMETRIES

Evaluations were performed to determine T_{enable} for other vessel geometries using the method described in this paper. In each case, additional operating margin can be obtained utilizing Equation 8 with the Code Case N-588 stress intensity factors for axial and circumferential flaws. The results of these evaluations are presented in Table 1.

Based on Table 1, a bounding set of T_{enable} values which envelopes all known PWR reactor configurations would be RT_{NDT} + 40°F for the axial flaw, and RT_{NDT} – 85°F for the circumferential flaw.

These T_{enable} values are based on the Code Case N-514 basis document definition of T_{enable} as that temperature at which the allowable pressure in the reactor vessel may reach 110% of the design pressure, without initiation of a quarter-thickness depth reference flaw. In this case, both the axial and circumferential directions are considered. While the values derived for circumferential flaws are in some cases substantially below RT_{NDT}, it should be noted that this is simply due to the lower stress intensity imposed on a circumferential flaw. The T_{enable} values derived still meet the fundamental Code Case N-514 basis document definition of T_{enable}.

CONCLUSION

This evaluation demonstrates the procedure for calculating T_{enable} on a plant specific basis using a methodology consistent with Appendix G of ASME Code Section XI. The procedure also provides consideration of alternate reference flaw orientation in accordance with Code Case N-588. This establishes T_{enable} such that an appropriate level of vessel protection against brittle failure is provided at low temperatures, while improving plant operating margins.

On this basis, allowing for a simplified bounding approach as well as an explicit plant-specific approach, ASME Section XI approved a Code Case to implement these procedures.

Table 1: Enable Temperature for Different Vessel Geometries

Vessel Type	Axial Flaw	Circumferential Flaw
Westinghouse 2-Loop	RT_{NDT} + 23°F	Any temperature
Westinghouse 3-Loop	RT_{NDT} + 30°F	RT_{NDT} – 174°F
Westinghouse 4-Loop	RT_{NDT} + 34°F	RT_{NDT} – 110°F
B&W 177-FA	RT_{NDT} + 35°F	RT_{NDT} – 103°F
Early CE Design	RT_{NDT} + 25°F	Any temperature
CE System 80	RT_{NDT} + 38°F	RT_{NDT} – 86°F

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