Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6650

W3F1-2001-0032
A4.05
PR

April 13, 2001

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Technical Specification Change Request NPF-38-230
Revision of RCS Cooldown Requirement

Gentlemen:

In accordance with 10CFR50.90, Entergy Operations, Inc. (EOI) is hereby proposing to amend Operating License NPF-38 for Waterford 3 by requesting the NRC Staff review and approval of the attached change to the Technical Specifications (TS). The proposed change relaxes the allowable cooldown rate in TS 3.4.8.1, "RCS Pressure / Temperature Limits". Specifically, this change will eliminate the limitation of a 10°F per hour cooldown rate when the RCS temperature is below 135°F. The proposed limitations permit a 100°F per hour cooldown rate to continue down to an RCS temperature of 110°F, at which point the rate is reduced to 30°F per hour. The attached description and safety analysis support the proposed change to the Waterford 3 TS. This change is desirable in order to improve outage efficiency.

This proposed change has been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and it has been determined that this request involves no significant hazards consideration.

The circumstances surrounding this change do not meet the NRC Staff criteria for exigent or emergency review. EOI requests review and approval of this amendment request on a schedule to support our upcoming outage in March, 2002. EOI requests the effective date for this TS change be within 60 days of approval.

A001

There are no commitments contained in this submittal. Should you have any questions or comments concerning this request, please contact Jerry Burford at (601) 368-5755.

Pursuant to 28 U.S.C.A. Section 1746, I declare under penalty of perjury that the foregoing is true and correct. Executed on April 13, 2001.

Very truly yours,

J.T. Herron
Vice President, Operations
Waterford 3

JTH/fgb/cbh

Attachments:
1. NPF-38-230, Technical Specification Change Request
2. NPF-38-230, Proposed Marked-Up Specifications
3. Westinghouse Report ER-WS-PS-0001, Rev. 000
4. NPF-38-230, Mark-up of Affected Technical Specification Bases Pages

cc:
E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
J. Smith
N.S. Reynolds
NRC Resident Inspectors Office
Administrator Radiation Protection Division
(State of Louisiana)
American Nuclear Insurers

ATTACHMENT 1

To W3F1-2001-0032

NPF-38-230

TECHNICAL SPECIFICATION CHANGE REQUEST
REVISION OF RCS COOLDOWN REQUIREMENT

DESCRIPTION AND NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION OF PROPOSED CHANGE NPF-38-230

Summary of Proposed Change

The proposed change revises the RCS cooldown requirements in Technical Specification (TS) 3.4.8.1, "RCS Pressure / Temperature Limits," to eliminate the 10°F per hour restriction and permit a 30°F per hour cooldown when the RCS temperature is below 110°F. The proposed change also extends the use of the 100 °F per hour cooldown rate down to an RCS temperature of greater than or equal to 110°F. This change will provide greater operational flexibility in the later stages of a plant cooldown. The proposed changes are supported by current plant analyses of the pressure / temperature (P/T) limits. The TS 3/4.4.8 Basis section is revised to describe the methods used to monitor P/T limits during various plant conditions.

In addition, as the proposed change frees space on page 3/4 4-28, it is proposed that the three lines from page 3/4 4-29 be moved up and that page shown as "intentionally left blank."

Proposed Marked-up Specification

See Attachment 2.

Background

Each licensee authorized to operate a nuclear power reactor is required by 10CFR50.36 to provide TS for the operation of the plant. In particular, 10CFR50.36(c)(2) requires that limiting conditions of operation be included in the TS. The Pressure / Temperature (P/T) limits, including allowable heatup and cooldown rates, are among the limiting conditions of operation represented in the TS. 10CFR50, Appendices G and H, describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting these P/T limits. An acceptable method for constructing the P/T limits is described in Standard Review Plan (NUREG-0800), Section 5.3.2.

Appendix G of 10CFR50 specifies fracture toughness requirements for reactor vessel materials based on the ASME Code. Demonstrating compliance with this appendix involves testing ferritic materials in accordance with the ASME Code and testing the vessel beltline materials in accordance with 10CFR50, Appendix H. Appendix H requires the establishment a surveillance program to periodically withdraw and test surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and contain test specimens made from plate, weld, and heat-affected-zone materials of the reactor beltline. ASTM E-185 provides additional details regarding the surveillance program and the specific testing required to address Appendix H. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in the materials' reference temperature. Appendix G also requires two supplemental fracture toughness tests to account for the effects of neutron irradiation on vessel embrittlement and Charpy upper shelf energy (USE). Generic Letter 88-11, "NRC Position On Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations," requested that licensees use the methods in Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the Adjusted Reference Temperature (ART) as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

By letter dated December 14, 1993, Entergy had proposed a TS change to extend the applicability basis of the P/T curves and limits from the period of 0 to 8 effective full-power years (EFPY) to 0 to 20 EFPY. The revised curves were based on ABB Combustion Engineering (CE) Report C-MECH-ER-021, Rev. 00 and Babcock & Wilcox (B&W) Report BAW-2177 (Analysis of Capsule W-97). This B&W report had been submitted for NRC Staff review in November of 1992.

The NRC Staff requested additional information by letter dated January 11, 1995. Specifically, there were questions concerning the methods for predicting fluence and the uncertainty associated with the measurements and calculations used in the fluence prediction. The specific concern was the projection of the fluence value and associated P/T limits to 20 EFPY, based on the analysis results of only one surveillance capsule. To resolve this issue, Entergy submitted a letter dated March 3,1995 requesting that the requested 20 EFPY be modified to 15 EFPY. The modified curves were still based on a 0 to 20 EFPY peak surface fluence of 2.29×10^{19} neutrons per square centimeter (n/cm2), but the applicability of the curves would simply be administratively limited to a shorter period. The NRC Staff issued Amendment 106 on May 18, 1995, approving the extended basis out to 15 EFPY.

Entergy again requested extension of the curve basis out to 20 EFPY by letter dated July 15, 1999. A new B&W report (51-1234900-00, Fluence Uncertainty Information For Extending Waterford Unit 3 P-T Limits to 20 EFPY) provided support that the fluence uncertainty was within the guidelines of Regulatory Guide 1.99. The requested extension was later reduced to 16 EFPY. This reduced request was found to be acceptable based on the Staff's estimation that the margin provided in the calculations that assumed 20 EFPY was sufficient to address the potential uncertainty associated with the determination of the fluence. The Staff utilized the methodology described in

Draft Regulatory Guide DG-1053 as the basis for their approval. Amendment 160 was issued on April 24, 2000 approving the use of the limits out to 16 EFPY.

The changes proposed in this submittal utilize the same CE Engineering Report referenced in the 1993 request as a basis for the current curves. Westinghouse Engineering Report ER-WS-PS-0001, Rev. 000, "Improvement of Technical Specification RCS Cooldown Limitations," re-assessed the information and determined there was sufficient margin to support revised cooldown limits. A copy of this report is provided as Attachment 3.

Report ER-WS-PS-0001, Rev. 000 also describes the most accurate methodology for monitoring reactor vessel beltline temperatures during various plant conditions. During reactor coolant pump operation, the lowest cold leg temperature associated with an operating reactor coolant pump is used to monitor P/T limits. When one or more of the reactor coolant pumps is running, cold leg temperature indication is representative of the coolant temperature entering the reactor vessel beltline. Following coastdown of the last RCP, the segments of reactor coolant piping upstream of shutdown cooling injection are no longer indicative of reactor beltline temperature. Therefore, shutdown cooling temperature is used to monitor P/T limits. Changes to the TS 3/4.4.8 Bases section are proposed to describe this method of monitoring reactor vessel beltline temperature.

Description and Safety Considerations

Waterford 3 proposes to amend its Operating License by modifying the Limiting Condition for Operation for Technical Specification (TS) 3.4.8.1 and Figure 3.4-3. The requested changes will revise the Reactor Coolant System (RCS) cooldown limitations to eliminate the 10 °F per hour restriction at RCS temperatures below 135 °F. The new cooldown rate steps are proposed as 100 °F per hour at RCS temperatures down to and including 110 °F and 30 °F per hour limit at RCS temperatures below 110 °F.

There are no changes proposed to either the applicability period for the curves nor to the low temperature overpressure protection (LTOP) enable temperature of 272 °F.

The attached Westinghouse report (attachment 3) provides the basis for the acceptability of the proposed changes. The report utilizes the existing data that serves as the basis for the current P/T limits. The data is adjusted to present actual rather than indicated pressure and temperature limit information referenced to the pressurizer pressure. This actual data is then used to assess whether the cooldown limits can be made less restrictive.

The basis for the acceptability of the minimum cooldown rate in this assessment is:

> The conservatively determined peak transient pressure must be less than the bounding pressure established for the applicable cooldown rate. This bounding pressure is that calculated minimum acceptable pressure for the limiting materials at temperatures ranging from the minimum boltup temperature (i.e., 72 °F indicated) and the LTOP enable temperature (272 °F indicated).

The bounding pressure is typically that at the minimum boltup temperature. As shown in Table 4 of the Westinghouse engineering report, those pressures are 571.9 psia, 513.9 psia, and 339.1 psia for cooldown rates of 10 °F, 30 °F, and 100 °F, respectively. The peak calculated transient pressure has been determined in existing calculations (referenced in the report) and conservatively adjusted in the report to be 465 psia. As can be seen from this data, the bounding pressure at the minimum boltup temperature for the 30 °F per hour cooldown rate is greater than the peak transient pressure; thus, a cooldown rate of 30 °F per hour is acceptable at all temperatures down to the minimum boltup temperature.

The data in Table 4 of the report is further evaluated to determine the lowest temperature at which the 100 °F per hour cooldown rate would be acceptable. The peak transient pressure of 465 psia is less than the actual pressurizer pressure limit at an actual RCS temperature of 70 °F. Thus, this cooldown rate, which is currently restricted by this Technical Specification to use above RCS temperatures of 135 °F (indicated), may be used down to an RCS temperature of 70 °F (actual). However, the technical specification limit is being conservatively established at 110°F (indicated) to account for instrument uncertainty.

Conclusion

The proposed changes to TS 3.4.8.1 will allow more operating flexibility during cooldown by permitting a higher cooldown rate at RCS temperatures below 135 °F. They have been demonstrated to be acceptable based on the comparison of the allowable pressurizer pressure limit over a range of RCS temperatures and ensuring that the minimum allowable pressure exceeds the conservatively calculated peak transient pressure condition.

No Significant Hazards Consideration Determination

Entergy Operations, Inc. is proposing that the Waterford 3 Operating License be amended to modify Technical Specification (TS) 3.4.8.1, "Reactor Coolant System Pressure/ Temperature Limits," as follows:

- Eliminate the 10 °F per hour cooldown limit currently at RCS temperatures below 135 °F.
- Extend the 100 °F per hour cooldown limit to be acceptable at RCS temperatures of 110 °F or greater.
- Establish a 30 °F per hour cooldown limit to be applicable at RCS temperatures below 110 °F.

The proposed changes described above have been evaluated in accordance with 10CFR50.92(c). The change shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will the operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response:

   Limitations have been imposed on cooldown of the Reactor Coolant System (RCS) to assure compliance with the minimum temperature requirements of 10CFR50, Appendix G. The proposed changes revise the allowable cooldown limits in a way such that operation remains consistent with the design assumptions and satisfies the stress limits for cyclic operation. By ensuring operation remains within the bounds of the existing design basis and assumptions, the probability of a brittle fracture of the reactor vessel has not been increased.

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

2. Will the operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response:

   The proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed since they do not introduce new systems, failure modes, or other plant perturbations. The proposed changes revise the cooldown limitations based on the fact the conservatively estimated peak pressure that can occur when the RCS cold leg temperature is below 200 °F is less than the proposed pressure limit. The limits assure that operation remains consistent with the design assumptions and satisfies the stress limits for cyclic operation.

Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will the operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response:

   The margin of safety provided by Technical Specification 3.4.8.1 is based on assuring that the maximum cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation. The proposed changes will not involve a significant reduction in the margin of safety since equivalent pressure and temperature limit requirements for reactor operation will be applied. The proposed changes were derived in accordance with approved NRC methodology which was developed to assure the reactor coolant system pressure boundary is designed with sufficient margin to withstand any condition during normal operation including anticipated operational occurrences and system in-service leak and hydrostatic tests.

   These requirements were revised in accordance with 10CFR50, Appendix G utilizing the latest NRC guidance in Regulatory Guide 1.99, Revision 2 relative to estimating neutron irradiation damage to the reactor vessel. In addition, the 16 EFPY basis for these pressure/temperature limits has been found to include sufficient margin to account for the limits of uncertainty described in Draft Regulatory Guide DG-1053.

   Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Conclusion

Based on the above No Significant Hazards Evaluation, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

ENVIRONMENTAL IMPACT EVALUATION

Pursuant to 10CFR51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR 51.22 (c) (9) of the regulations. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. This change does not result in a significant change or significant increase in the radiological doses for any design basis accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because the proposed change does not modify any equipment or alter the way equipment operates.

ATTACHMENT 2

To W3F1-2001-0032

NPF-38-230

PROPOSED MARKED-UP SPECIFICATIONS

REACTOR COOLANT SYSTEM

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 30°F per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of 50°F per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of 60°F per hour with Reactor Coolant System cold leg temperature greater than 345°F.

~~d. A maximum cooldown rate of 10°F per hour with Reactor Coolant System cold leg temperature less than 135°F.~~

d ~~e~~. A maximum cooldown rate of 30°F per hour with Reactor Coolant System cold leg temperature ~~greater than or equal to 135°F and~~ less than ~~or equal to 200°F.~~ 110°F

e ~~f~~. A maximum cooldown rate of 100°F per hour with Reactor Coolant System cold leg temperature greater than ~~200°F.~~ or equal to 110°F

f ~~g~~. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the

SURVEILLANCE REQUIREMENTS (Continued)

Reactor Vessel material surveillance program - withdrawal schedule in FSAR Table 5.3-10. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

- move text to preceding page (space permitting)
- replace with: "This page intentionally left blank"
- remove header information

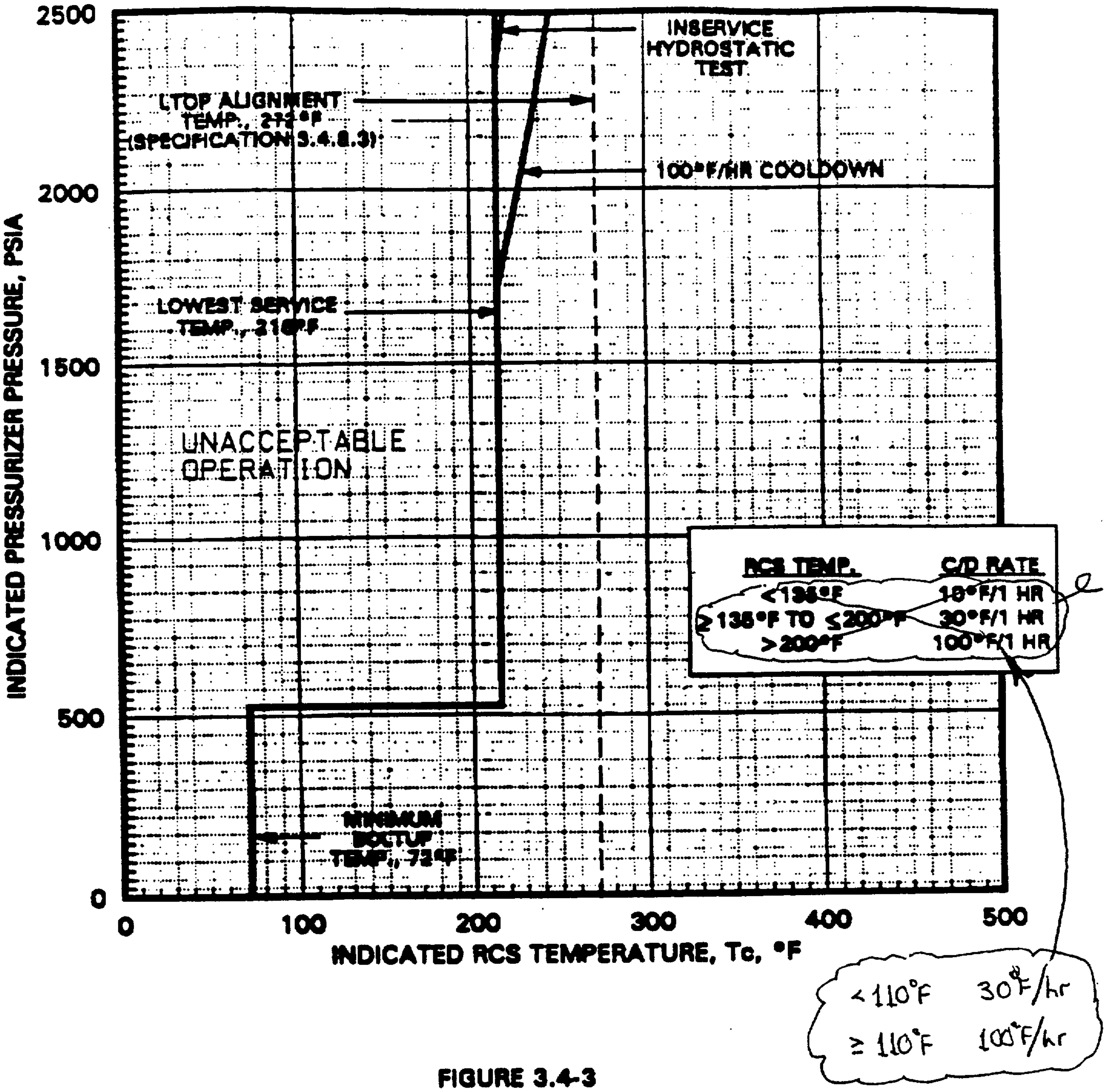

FIGURE 3.4-3

WATERFORD UNIT 3 COOLDOWN CURVE

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS

0-16 EFPY

CURVE BASIS: PEAK SURFACE FLUENCE = 2.29×10^{19} n/cm^2 @ 20 EFPY

ATTACHMENT 3

To W3F1-2001-0032

NPF-38-230

WESTINGHOUSE REPORT ER-WS-PS-0001, REV 000
IMPROVEMENT OF TECHNICAL SPECIFICATION RCS COOLDOWN LIMITATIONS

ENTERGY OPERATIONS, INC.
WATERFORD STEAM ELECTRIC STATION UNIT 3

IMPROVEMENT OF TECHNICAL SPECIFICATION RCS COOLDOWN LIMITATIONS

ENGINEERING REPORT

ER-WS-PS-0001, REV. 000

Prepared by: ______________________ **Date:** 6.30.00

A. A. Ostrov

VERIFICATION STATUS: COMPLETE

The design information contained in this document has been verified to be correct by means of Design Review

Reviewed by: ______________________ **Date:** 6/30/2000

R. M. Orsulak, Independent Reviewer

Approved by: ______________________ **Date:** 6/30/2000

C. J. Gimbrone, Supervisor, Plant Systems

Pages: 25 total

WESTINGHOUSE NUCLEAR SERVICES

TABLE OF CONTENTS

1 INTRODUCTION

Plant cooldown at Waterford Steam Electric Station Unit 3 (WSES-3) is controlled by the following Technical Specification (TS) RCS cooldown (CD) limitations (Reference 6.1, LCO 3.4.8.1):

RCS Temp., °F	CD Rate, °F/hr
< 135	10
135 - 200	30
> 200	100

As these limitations became too restrictive for operation, especially at the lower RCS temperatures, Entergy Operations, Inc. (Entergy) had initiated an effort with an objective of assessing the possibility of their improvement. Entergy determined that a complete elimination of the existing 10°F/hr CD rate with a simultaneous extension of the applicability of the 30°F/hr CD rate from the current 135°F to the minimum boltup temperature (72°F) would be a desirable improvement. (Note that the above temperatures are indicated.)

That effort resulted in a best estimate evaluation, which determined that sufficient margin existed in the current analyses that would allow achieving the objective. As that evaluation, Reference 6.2, was just an assessment and has not been performed by ABB (now Westinghouse Nuclear Services) as a safety related activity, it lacked a pedigree of a supporting document to a license submittal on an improvement of the CD limitations. Subsequently, Reference 6.3 requested Westinghouse Nuclear Services to perform a new, safety-related evaluation that contains sufficient detail to support the license submittal. The subject report documents this new evaluation.

The report also contains base P-T limits at the reactor vessel beltline in terms of actual pressurizer pressure and actual reactor coolant temperature, i. e., without instrumentation uncertainties. These actual P-T limits for heatup, cooldown, and hydrostatic test are obtained from the existing indicated P-T limits for 20 EFPY by removing the existing pressure and temperature instrumentation uncertainties. These actual base limits are intended to be used for the development of future indicated P-T limits should pressure and temperature instrumentation uncertainties change. Guidance on the development of indicated P-T limits is also provided. The actual P-T limits are used here as a basis for the improvement in the CD limitations.The report also recommends the instruments that should be used for monitoring RCS pressure and temperature during RCS heatup, cooldown, and shutdown operations.

The report was prepared as a safety-related, Quality Class 1 document in accordance with the company's Quality Procedures Manual QPM-101.

2 METHOD

The subject evaluation is based on two major inputs: actual RCS pressure-temperature (P-T) limits and a bounding value for the peak transient pressure. The existing P-T limits in TS are presented in terms of indicated pressurizer pressure and indicated cold leg temperature, whereas the P-T limits to be used as a basis in this evaluation should be in terms of actual pressurizer pressure and actual cold leg temperature, as requested in Reference 6.3. Accordingly, the first step in the evaluation is to modify the indicated pressure correction factors (IPCF), which were applied to the reactor vessel beltline (base) P-T limits to obtain the TS P-T limits, to the actual pressure correction factors (APCF) by removing uncertainties associated with pressurizer pressure indication instrumentation loops. This task is addressed in Section 3.1.

Next, the existing indicated P-T limits are modified by replacing the IPCFs by the APCFs obtained in Section 3.1 and removing the uncertainty associated with cold leg temperature indication. The resulting actual P-T limits are then used as a basis for the determination of the new CD limitations. This task is addressed in Section 3.2. Section 3.2 also contains guidance on a reverse process, i. e., application of pressure and temperature indication uncertainties to the actual P-T limits to obtain indicated P-T limits, as requested in Reference 6.3.

Section 3.3 documents a determination of a bounding value for the peak transient pressure that allows meeting the objective stated in Section 3.1, which is a complete elimination of 10°F/hr CD rate and extension of the applicability of 30°F/hr CD rate to the minimum boltup temperature. This section also determines and justifies a conservative value for the peak transient pressure via review of the inputs and assumptions utilized in the existing transient analyses.

Attachment 1, Section 5.0 of Reference 6.2 concluded that reanalyses of the existing mass and energy addition transients with updated inputs and assumptions would either yield insignificant increases in the existing peak transient pressures, due to a large size of the LTOP relief valves, or actually reduce the existing peak transient pressures, due to use of more realistic assumptions. The evaluation contained in Section 3.3 justifies a conservative value for the peak transient pressure without a detailed reanalysis. It is also demonstrated that the justified value is less than the bounding value, which ensures that the objective of the evaluation is met. The justified value for the peak transient pressure is then used as a basis for the determination of the new CD limitations.

The final task of the evaluation is the determination of the new CD limitations, based upon the P-T limits of Section 3.2 and the peak transient pressure of Section 3.3. The criterion used in the determination of the CD limitations is that the P-T limits for the selected CD rates are maintained above the peak transient pressure. This meets the objective of the low temperature overpressure protection (LTOP) system, which is to preclude violation of P-T limits during the worst-case pressure transient, as stated in Reference 6.4. Note that Reference 6.4 is used by the NRC in evaluations of submittals related to LTOP. This task is documented in Section 3.4.

The report also documents review of the applicable WSES-3 operating procedures and identifies the pressure and temperature instruments that should be used for monitoring RCS pressure and temperature during RCS heatup, cooldown, and shutdown operations. This task is included as requested in Reference 6.3.

3 EVALUATION

3.1 ACTUAL PRESSURE CORRECTION FACTORS

This section determines the actual pressure correction factors (APCF) from the existing indicated pressure correction factors (IPCF) of Reference 6.5. The existing IPCFs are 100 psi for narrow range pressure indication and 186 psi for wide range pressure indication. Each IPCF includes the following common terms:

- Elevation head, ΔP_Z = 36.04 psi
- Hot leg flow induced pressure drop, ΔP_{HL} = 0.196 psi
- Reactor vessel flow induced pressure drop, ΔP_{RV} = 34.71 psi,

for a total of 36.04 + 0.196 + 34.71 = 70.946 psi.

Additionally, each IPCF includes pressure instrumentation uncertainty associated with the appropriate control room indication loop. The uncertainty for the narrow range loop is ± 28.34 psi, which yields the $IPCF_{NR} = \Delta P_{NR\,CORR} = 70.946 + 28.34 = 99.286$ psi, or a rounded up value of 100 psi. The uncertainty for the wide range loop is ± 114.89 psi, which yields the $IPCF_{WR} = \Delta P_{WR\,CORR} = 70.946 + 114.89 = 185.836$ psi, or a rounded up value of 186 psi. Note that for an IPCF, a conservative value for uncertainty is with the "plus" sign.

Without the instrumentation uncertainties, there is only one APCF that is to be used for the actual P-T limits, i. e., 70.946 psi, or 71 psi, when rounded up.

3.2 PRESSURE-TEMPERATURE LIMITS

This section determines the actual reactor vessel P-T limits at the beltline (base P-T limits) and in the pressurizer. The actual Minimum Pressure value, calculated in Reference 6.6, is also determined. Finally, the section provides guidance on obtaining the indicated P-T limits from the actual P-T limits.

3.2.1 Actual P-T Limits

Tables 3, 4, and 5 of Reference 6.6 provide the applicable indicated P-T limits for 20 EFPY for heatup, cooldown, and hydrostatic test, respectively. The RCS temperature values in the tables were obtained by adding temperature uncertainty of 25.6°F to the assumed actual RCS (cold leg) temperatures (Reference 6.6, pg. 11). To arrive at the corresponding actual temperatures, this uncertainty is subtracted from the table temperature values.

The pressure values in the tables were obtained by subtracting the IPCFs from the base P-T limit pressure values that were calculated for each assumed temperature. To arrive at the corresponding actual base pressure values, the following IPCFs are added back to the table pressure values (Reference 6.6, pg.11):

Actual Pressure (P) Range	Total Pressure Correction Factor (PCF)
P< 200 psia	186 psi
200 psia ≤ P< 850 psia	100 psi
850 psia ≤ P< 3000 psia	186 psi

The resulting base P-T limits are summarized in Tables 1, 2, and 3.

To obtain the actual P-T limits in terms of pressurizer pressure, APCF = 71 psi (see Section 3.1) should be added back to the pressure values of Tables 1, 2, and 3. As the scope of the subject evaluation is limited to improvement of the cooldown limitations, the actual P-T limits in terms of pressurizer pressure are only determined for the cooldown values in Table 2. These actual P-T limits in the pressurizer are tabulated in Table 4. Table 4 data, along with the Minimum Pressure value as described below, provides a basis for the determination of the new cooldown limitations.

3.2.2 Minimum Pressure

The composite P-T limits of Reference 6.1, as well as those in Reference 6.6, contain other limitations in addition to the reactor vessel beltline limits. One of these additional limitations is the Minimum Pressure, as described in Section 2.8 of Reference 6.6. The Minimum Pressure is an upper pressure limit in the region between the minimum boltup temperature (72°F indicated) and the Lowest Service Temperature (216°F indicated) that cannot be exceeded during a postulated worst-case overpressure transient. The value has the same significance as the beltline P-T limits, although it applies to the entire reactor coolant pressure boundary. The Minimum Pressure needs to be accounted for in this temperature region to determine whether it is more restrictive (less) than the applicable beltline P-T limits.

Note that the minimum boltup temperature including temperature instrumentation uncertainty (25.6°F) was determined to equal 45.6°F, which was then increased to 72°F for additional conservatism (Reference 6.6, pg. 19). The latter is included in the TS (see Figures 3.4-2 and 3.4-3 of Reference 6.1). Using the TS value, the actual minimum boltup temperature is thus 46.4°F (72°F-25.6°F).

The actual value for the Lowest Service Temperature was calculated to equal 190°F (Reference 6.6, pg. 18). It was then increased by temperature instrumentation uncertainty (25.6°F) and rounded up to 216°F. Thus the applicability of the Minimum Pressure is between 46.4°F and 190°F actual.

The existing uncorrected value for the Minimum Pressure is 625 psia, per Reference 6.6, Section 2.8. Applying the same APCF=71 psi, as that used for the beltline limits, the Minimum Pressure in terms of actual pressurizer pressure is 625 psia -71 psi = 554 psia. This value will be compared with the applicable beltline limits of Table 4 to determine the limiting (lowest) value.

3.2.3 Guidance for Development of Indicated P-T Limits

There are two approaches to developing indicated reactor vessel beltline P-T limits from actual P-T limits, depending on the reference location of the actual P-T limits. If the actual P-T limits are referenced to the beltline (base limits), an indicated pressure correction factor (IPCF) must be subtracted from each actual pressure value at the beltline. This is a reverse process to the one described in Section 3.1. Typically, there are two IPCFs, one for each the narrow and wide range pressurizer pressure indication channel. This approach would be valid with the values in Tables 1, 2, and 3.

Another approach should take place if the actual P-T limits are already adjusted to the pressurizer, as in Table 4. In this case, only pressurizer pressure indication loop uncertainty (-ies) should be subtracted from the actual P-T limits to arrive at the indicated P-T limits.

In both cases, the most limiting reactor coolant temperature indication uncertainty should be added to the actual coolant temperatures, which are assumed to generate a pressure vs. temperature function. See Section 4 for further clarification.

TABLE 1. BASE P-T LIMITS AT THE BELTLINE: HEATUP, 20 EFPY

(Instrumentation uncertainties are not included)

Actual RCS Temperature, °F	Actual RCS Pressure, psia			
	Isothermal	30°F/hr	50°F/hr	60°F/hr
46.4	672.0	672.0	672.0	672.0
50.0	682.0	682.0	682.0	682.0
60.0	710.8	710.8	710.8	710.8
70.0	744.1	718.4	698.9	694.5
80.0	782.6	732.7	694.0	683.5
90.0	827.1	759.8	701.4	683.7
94.4	849.9*	-	-	-
94.5	850.0*	-	-	-
100.0	878.6	792.6	719.9	694.5
110.0	938.1	846.8	748.7	715.1
110.5	-	849.9*	-	-
110.6	-	850.0*	-	-
120.0	1006.9	905.9	787.6	745.5
130.0	1086.4	975.7	836.9	785.5
132.4	-	-	849.9*	-
132.5	-	-	850.0*	-
140.0	1178.4	1057.9	897.1	836.1
142.4	-	-	-	849.9*
142.5	-	-	-	850.0*
150.0	1284.6	1153.3	969.2	897.6
160.0	1407.5	1264.5	1054.6	971.7
170.0	1549.5	1392.9	1154.8	1059.0
174.3	-	1457.0*	-	-
174.4	-	-	1206.3*	-
180.0	1713.7	1542.0	1271.9	1162.1
190.0	1903.5	1714.0	1408.2	1282.2
200.0	2123.0	1913.6	1566.5	1422.8
202.4	-	-	1610.6*	-
210.0	2376.7	2143.6	1750.2	1585.7
220.0	2669.9	2410.5	1963.0	1775.5
220.6	2686.0*	-	-	-
230.0	-	2709.5	2209.4	1994.8
240.0	-	-	2494.5	2249.8
245.8	-	-	2686.0*	-
250.0	-	-	2824.5	2544.0
260.0	-	-	-	2885.7

NOTE: * Interpolated value.

TABLE 2. BASE P-T LIMITS AT THE BELTLINE: COOLDOWN, 20 EFPY
(Instrumentation uncertainties are not included)

Actual RCS Temperature, °F	Actual RCS Pressure, psia			
	Isothermal	10°F/hr	30°F/hr	100°F/hr
40.0	-	626.0	565.7	380.5
46.4	672.0	642.9*	584.9*	410.1
50.0	682.0	652.4	595.7	426.7
60.0	710.8	683.2	630.4	480.0
70.0	744.1	718.5	670.4	541.7
80.0	782.6	759.6	716.9	613.0
90.0	827.1	806.8	770.3	695.3
94.4	849.9*	-	-	-
94.5	850.0*	-	-	-
98.0	-	849.9*	-	-
98.1	-	850.0*	-	-
100.0	878.6	861.7	792.6	790.6
102.6	-	-	849.9*	-
102.7	-	-	850.0*	-
105.5	-	-	-	849.9*
105.6	-	-	-	850.0*
109.3	-	920.4*	-	-
109.4	-	-	899.4*	-
110.0	938.1	924.8	903.8	900.6
120.0	1006.9	998.2	986.8	1006.9
130.0	1086.4	1082.6	1082.2	1086.4
140.0	1178.4	1178.4	1178.4	1178.4
150.0	1284.6	1284.6	1284.6	1284.6
160.0	1407.5	1407.5	1407.5	1407.5
170.0	1549.5	1549.5	1549.5	1549.5
174.4	-	-	1621.7*	-
174.5	-	-	-	1623.4*
180.0	1713.7	1713.7	1713.7	1713.7
190.0	1903.5	1903.5	1903.5	1903.5
200.0	2123.0	2123.0	2123.0	2123.0
210.0	2376.7	2376.7	2376.7	2376.7
220.0	2669.9	2669.9	2669.9	2669.9
220.6	2686.0*	2686.0*	2686.0*	2686.0*

NOTE: * Interpolated value.

TABLE 3. BASE P-T LIMITS AT THE BELTLINE: HYDROSTATIC TEST, 20 EFPY

(Instrumentation uncertainties are not included)

Actual RCS Temperature, °F	Actual RCS Pressure, psia
46.4	896.2
50.0	909.3
60.0	947.7
70.0	992.1
80.0	1043.5
90.0	1102.8
100.0	1171.5
110.0	1250.8
120.0	1342.5
130.0	1448.6
140.0	1571.1
150.0	1712.8
160.0	1876.7
170.0	2066.0
180.0	2285.0
190.0	2538.1
194.2	2661.0*
195.1	2686.0*
200.0	2830.6

NOTE: * Interpolated value.

TABLE 4. ACTUAL P-T LIMITS IN THE PRESSURIZER: COOLDOWN, 20 EFPY

(Instrumentation uncertainties are not included)

Actual RCS Temperature, °F	Actual Pressurizer Pressure, psia			
	Isothermal	10°F/hr	30°F/hr	100°F/hr
40.0	-	555.0	494.7	309.5
46.4	601.0	571.9*	513.9*	339.1
50.0	611.0	581.4	524.7	355.7
60.0	639.8	612.2	559.4	409.0
70.0	673.1	647.5	599.4	470.7
80.0	711.6	688.6	645.9	542.0
90.0	756.1	735.8	699.3	624.5
94.4	778.9*	-	-	-
94.5	779.0*	-	-	-
98.0	-	778.9*	-	-
98.1	-	779.0*	-	-
100.0	807.6	790.7	721.6	719.6
102.6	-	-	778.9*	-
102.7	-	-	779.0*	-
105.5	-	-	-	778.9*
105.6	-	-	-	779.0*
109.3	-	849.4*	-	-
109.4	-	-	828.4*	-
110.0	867.1	853.8	832.8	829.6
120.0	935.9	927.2	915.8	935.9
130.0	1015.4	1011.6	1011.2	1015.4
140.0	1107.4	1107.4	1107.4	1107.4
150.0	1213.6	1213.6	1213.6	1213.6
160.0	1336.5	1336.5	1336.5	1336.5
170.0	1478.5	1478.5	1478.5	1478.5
174.4	-	-	1550.7*	-
174.5	-	-	-	1552.4*
180.0	1642.7	1642.7	1642.7	1642.7
190.0	1832.5	1832.5	1832.5	1832.5
200.0	2052.0	2052.0	2052.0	2052.0
210.0	2305.7	2305.7	2305.7	2305.7
220.0	2598.9	2598.9	2598.9	2598.9
220.6	2615.0*	2615.0*	2615.0*	2615.0*

NOTE: The pressures are obtained from RCS pressure values of Table 2 by subtracting APCF =71 psi, except for the pressures marked by an asterisk (*), which are interpolated values.

3.3 PEAK TRANSIENT PRESSURES

This section determines a bounding value for the peak transient pressure that allows meeting the objective stated in Section 3.1 and determines a conservative value for the peak transient pressure via review of the existing inputs and assumptions.

3.3.1 Bounding Peak Pressure

In order to eliminate 10°F/hr CD rate and extend the applicability of 30°F/hr CD rate to the minimum boltup temperature (46.4°F actual), the peak transient pressure cannot exceed the most limiting (lowest) pressure on the 30°F/hr CD curve within the LTOP temperature range. The LTOP temperature range is between the minimum boltup temperature (see above) and the LTOP enable temperature. The latter, 272°F indicated (Reference 6.1, LCO 3.4.8.3) , includes temperature uncertainty of 25.6°F. As the lowest P-T limit is typically found at the minimum boltup temperature, this and the adjacent temperatures are herein considered.

The peak transient pressure must also be less than the Minimum Pressure of 554 psia (Section 3.2.2).

Per Table 4, the actual pressurizer pressure for 30°F/hr CD at the minimum boltup temperature of 46.4°F is 513.9 psia. As this value is less than the Minimum Pressure of 554 psia, the 513.9 psia can be considered as a bounding value for the bounding peak pressure. To provide a margin, a value of 510 psia is selected as the bounding peak pressure that allows meeting the task's objective.

3.3.2 Review of Transient Analyses

The existing peak transient pressure is 460 psia in both the mass addition and energy addition transients, per References 6.7 and 6.8. This value is equal to the maximum pressure at the valve inlet when the valve first opens. The valve opening characteristics used in the transient analyses included valve initial opening at 7% accumulation above the set pressure, at which point the valve pops open to 70% of its full opening and reaches 70% of its rated capacity. This capacity is more than sufficient to relieve each pressure transient, with the peak pressure limited to 7% accumulation, i. e., (430 psia - 14.7 psi) x 1.07 +14.7 psi = 459.1, or 460 psia.

After reaching 70% of its full opening, the valve continues its opening at a ramp rate until it reaches full open position at 10% pressure accumulation above the set pressure, at which point it reaches its rated capacity. If inlet pressure continues rising, so does the flow; however, this portion of the opening characteristic is irrelevant for the subject evaluation, as valve capacity even at a partial opening is more than sufficient to relieve both these transients.

The valve rated capacity at 10% pressure accumulation above the set pressure of 430 psia, i. e., at (430 psia - 14.7 psi) x 1.1 +14.7 psi = 471.5 psia, is 3,089 gpm, per Reference 6.9. Accordingly, valve capacity at 7% accumulation is 0.7 x 3,089 gpm = 2,162 gpm. For comparison, flow rate from two HPSI pumps at a lower pressure of 392 psia (more conservative) is 1450 gpm, per Reference 6.7, Table 8.1.1. When combined with flow rate from three charging pumps (132 gpm), the total volumetric input into the RCS during the mass addition transient is approximately 1,600 gpm, which is less than 75% of the valve's relief capability at 7% accumulation.

Review of the analysis inputs and assumptions documented in Reference 6.2, Attachment 1, indicated some inconsistency with the current plant operations and LTOP transient analysis methods. However, the changes recommended there for revisions to the transient analyses will not appreciably increase the peak pressure, due to a significant size of the LTOP relief valve. Based upon data provided above, it would take 2,162 -1,600 = 562 gpm of additional flow rate to arrive at the equilibrium with valve capacity.

Furthermore, two factors used in the existing pressure transient analyses are actually overly conservative by today's methods. These are discussed in detail in the following section.

3.3.3 Estimation of Conservative Peak Pressure

Two overly conservative factors used in the existing pressure transient analyses can be reconsidered, which will reduce the peak transient pressure in the pressurizer. These factors are the valve opening characteristic and an elevation difference between the location of the valve and the pressurizer reference point that is used for normalizing the P-T limits.

The assumption of the valve initial opening at 7% accumulation is much more limiting than that allowed in the ASME Code. Per para. 2.5.5 of Reference 6.2, Attachment 1, this model has been modified to be consistent with ASME Code requirements to spring relief valves. Specifically, the model assumes valve initial opening at 3% accumulation, with capacity at 30% of the rated capacity, with continued opening at a ramp rate until 10% accumulation and rated capacity are reached. It is easy to determine the valve position and the eqilibrium pressure for the above mass addition transient:

$$1{,}600 \text{ gpm} = (\% \text{ Capacity})\ (3{,}089 \text{ gpm}),$$

Or,

$$(\% \text{ Capacity}) = (1{,}600 \text{ gpm}/3{,}089 \text{ gpm})(100\%) = 52\%,$$

which occurs at 5.2% pressure accumulation above the set pressure:

$$(430 \text{ psia} - 14.7 \text{ psi})(1.052) + 14.7 \text{ psi} = 451.6 \text{ psia},$$

which is less than the 460 psia peak pressure value calculated based on opening at 7% accumulation.

The second factor used in the determination of the peak pressure is an assumption that the RCS, including the pressurizer, is represented as a single node. It is a conservative assumption, which treats the LTOP relief valve and the pressurizer as being at the same elevation. As a result, the 460 psia peak pressure at the valve inlet is also considered to be pressurizer pressure, whereas a corresponding pressurizer pressure is less.

Thus, with the operating pressurizer water level elevation at 43.605 ft and the relief valve elevation at 22.5 ft, Reference 6.9 calculates the pressure differential as follows (see pg. 6 there):

$$(43.605 \text{ ft} - 22.5 \text{ ft})(0.3866 \text{ psi/ft}) = 8.2 \text{ psi}$$

This is a conservative value with respect to pressure adjustment, because 43.605 ft is not based on a water-solid pressurizer, which is one of the key assumptions in the transient analyses.

Based on the discussion presented above and for the same inputs and assumptions used in the transient analyses, the peak transient pressure in the pressurizer is around 451.6 psia - 8.2 psi = 443.4 psia, which would occur at an approximately 50% valve opening.

Considering the fact that the scope does not include determination of the peak pressure through analysis, a safety margin is incorporated into the final peak pressure value. Instead of using the 443.4 psia peak pressure determined above, it is conservatively assumed that to mitigate the transients, the valve actually reaches the full open position at 10% pressure accumulation, with the peak pressure calculated as follows (see Section 3.3.2):

$$(430 \text{ psia} - 14.7 \text{ psi}) \times 1.1 + 14.7 \text{ psi} = 471.5 \text{ psia}$$

Subtracting the pressure difference due to elevation head, 8.2 psi, the final conservative peak pressure value is thus 471.5 psia - 8.2 psi = 463.3 psia, or 465 psia, when rounded up.

3.4 REVISED COOLDOWN LIMITATIONS

Per Section 3.3.1, a conservatively determined peak transient pressure must be less than the bounding transient pressure value, 510 psia, to meet the objective of this effort. The conservatively estimated peak pressure value, 465 psia (Section 3.3.3), is actually less than 510 psia. Thus the main objective of the effort is met. Specifically, 10°F/hr cooldown rate can be completely eliminated and 30°F/hr cooldown rate can be used down to the minimum boltup temperature.

The next step is to determine the coolant temperature, above which cooldown rate of 100°F/hr can be used. Currently, this temperature is 200°F , as indicated in Section 1.

Per Table 4, the actual pressurizer pressures above and below the new peak pressure of 465.0 psia for 100°F/hr cooldown rate are 470.7 psia at 70.0°F and 409.0 psia at 60°F. Assuming a linear function between these two points, the P-T limit temperature that corresponds to the peak pressure of 465 psia is calculated as follows:

$$(P_{70} - P_{60}, \text{psi})/(P_{70} - 465, \text{psi}) = (70 - 60, {}^\circ\text{F})/(70 - t_{465}, {}^\circ\text{F})$$

$$(470.7 - 409.0)/(470.7 - 465) = (10)/(70 - t_{465})$$

$$61.7/5.7 = (10)/(70 - t_{465})$$

$$70 - t_{465} = (5.7 \times 10)/(61.7)$$

$$t_{465} = 69.1^\circ\text{F}$$

The value is conservatively rounded up to 70°F. Thus above the RCS cold leg temperature of 70°F (actual), the 100°F/hr cooldown curve is above the peak transient pressure of 465 psia. This signifies the fact that above 70°F (actual), the 100°F/hr cooldown curve is protected from being exceeded during the worst-case overpressure event. As a result, the applicability of the maximum allowable RCS cooldown rate of 100°F/hr can be extended down to t_c just above 70°F (actual). At $t_c \leq 70^\circ$F (actual), the maximum allowable cooldown rate must be reduced from 100°F/hr to 30°F/hr.

4 INSTRUMENTATION REQUIREMENTS

4.1 DISCUSSION

There are a number of TS requirements that are aimed at protection of the reactor coolant pressure boundary and, especially, the reactor vessel from brittle fracture. Brittle fracture may occur at low RCS temperatures, if parameters in the P-T limit and LTOP-related Technical Specifications are beyond the allowable range. Surveillance requirements in TS and operating procedures are designed to ensure that the relevant parameters, most notably, RCS pressure and temperature, are within the TS limits.

Compliance with P-T limits and LTOP requirements is implemented via adhering to operating procedures that instruct the operators to follow procedural steps related to monitoring and/or recording RCS pressure and temperature. A sample of these steps is provided below. The listing is not all inclusive. Similar steps can also be found in other procedures.

<u>From Reference 6.10</u>:

- Maintain the RCS within the TS P-T limits. Steps 3.1.1, 9.1.10, 9.2.8;
- Maintain RCS pressure below 392 psia with shutdown cooling in service. Step 3.2.3;
- Adhere to a 30°F/hr heatup rate limit below 200°F. Step 9.1.37;
- Record RCS heatup rate every 15 minutes on the RCS Heatup Log. Steps 9.1.38, 9.2.3;
- Adhere to 30°F/hr and 50°F/hr heatup limits. Steps 9.2.3, 9.2.21;
- Remove LTOP from service when all RCS cold leg temperatures are > 272°F. Step 9.2.19.4.2.

<u>From Reference 6.11</u>:

- Maintain the RCS within the TS P-T limits. Steps 3.1.1, 9.2.10, 9.2.18, 9.2.22;
- Record RCS cooldown rate every 15 minutes on the RCS Cooldown Log. Step 9.2.1;
- Adhere to a maximum CD rate of 100°F/hr above 200°F. Steps 9.2.10, 9.3.6, 9.4.4;
- Align the SDC suction relief valves for LTOP at ≤ 272°F. Steps 9.3.2, 9.4.2;
- Adhere to a maximum CD rate of 30°F/hr between 135 - 200°F. Steps 9.3.6, 9.3.19, 9.4.4, 9.4.15.

<u>From Reference 6.12</u>:

- Don't initiate shutdown cooling until RCS temperature <350°F and RCS pressure <392 psia. Step 3.2.1;
- Limit RCS temperature and pressure in accordance with TS. Step 3.2.2;
- Provide LTOP when any RCS cold leg temperature is ≤ 272°F. Step 3.2.4;
- Don't exceed reactor coolant cooldown rates. Step 6.1.11, 6.2.11.

The following sections contain recommendations on the appropriate pressure and temperature instruments to be used by the control room operators to monitor the RCS pressure and temperature to prevent violation of the P-T limits and LTOP requirements.

4.2 RCS PRESSURE INSTRUMENTATION

Pressurizer pressure instrumentation channels with control room indication/recording are used for monitoring RCS pressure. The wide range channels should cover the entire pressure range of the P-T limits, i. e., approximately, 300 - 2500 psia indicated. The narrow range channels should be sufficient for LTOP conditions, due to a low pressure limit for Shutdown Cooling System alignment (392 psia indicated) and low LTOP relief valve set pressure (430 psia at the valve). As each channel has its own accuracy, uncertainties should be calculated and each channel selected for pressure monitoring should be accompanied with its uncertainty, so the operators know the associated indication error.

4.3 REACTOR COOLANT TEMPERATURE INSTRUMENTATION

For the purposes of monitoring P-T limits and LTOP parameters, the lowest cold leg temperature entering the reactor vessel is used. The selected channels should have control room indicators/recorders. The cold leg temperature indications in all operating loops (RCP in operation) should be noted and compared to each other to select the lowest value. The selected value is then used as appropriate in the procedural steps such as those listed in Section 4.1.

When one or more RCPs are operating concurrent with operation of a SDCS (LPSI) pump, the cold leg temperature indications remain representative of the coolant temperature entering the reactor vessel beltline region. This is due to mixing between the large amount of the RCP-driven coolant, about 100,000 gpm per operating cold leg, with a relatively small quantity of the LPSI-driven coolant (about 4,000 gpm) entering that leg, even if the temperatures are different.

Following coastdown after the last RCP has been secured, the portions of reactor coolant on the RCP/SG sides of the SDC nozzles in the cold legs will most likely remain unaffected by the SDC flow. As a result, cold leg temperature indications may not be representative of the coolant temperature entering the reactor vessel. (The cold leg temperature elements are located between the SDC nozzles and RCPs.) During this configuration, the indications from LPSI Pump A (or B, or both, as applicable) Discharge Header Temperature Indicator, such as SI-ITI-0351X (SI-ITI-0352X), per Reference 6.12, should be used for coolant temperature indication.

As temperature readings are taken every 15 minutes, there could be an unusually large difference between the last reading from the cold leg temperature indicators while RCPs are operating, and the first reading from the LPSI pump discharge temperature indicators after RCP coastdown. This may result in an abnormal cooldown rate that exceeds the maximum allowable. To minimize this impact, it is recommended that prior to securing the last RCP(s), LPSI pump discharge temperature is raised to more closely match cold leg temperature.

Similar to the pressure channels, uncertainties for the temperature channels should be calculated and each channel selected for cold leg temperature monitoring should be accompanied with its uncertainty, so the operators know the associated indication error.

5 SUMMARY

5.1 Actual reactor vessel pressure-temperature (P-T) limits at the beltline location are developed (base P-T limits) from the existing indicated P-T limits at the pressurizer reference location (see Tables 1, 2, and 3). No adjustments and pressure/temperature uncertainties are included.

5.2 Actual reactor vessel P-T limits at the pressurizer reference location are developed for cooldown only to serve as a basis in the evaluation that determines revised cooldown limitations (see Table 4). No pressure/temperature uncertainties are included.

5.3 Guidance is provided on a conversion of the actual P-T limits of item 5.1 to indicated P-T limits at the pressurizer (see Section 3.2.3).

5.4 A conservative value for the peak transient pressure is estimated. The value is 465 psia in the pressurizer. It is significantly less than the bounding peak pressure value, 510 psia, which allows meeting the evaluation objective per Section 1.

5.5 Existing analysis margins allow for a significant improvement in the Technical Specification RCS cooldown limitations. Specifically, the three cooldown rates of 10°F/hr, 30°F/hr, and 100°F/hr are replaced by only two rates: 30°F/hr and 100°F/hr. The lower temperature limit for 100°F/hr cooldown, 200°F indicated, is extended down to 70°F actual, which translates to 95.6°F indicated if the existing temperature instrumentation indication uncertainty of 25.6°F is applied.

At the RCS temperatures ≤70°F actual, the maximum allowable cooldown rate is 30°F/hr. For plant operation it means that 30°F/hr cooldown rate will rarely be used, as the LTOP condition during cooldown is terminated when the reactor vessel head is lifted, which occurs at the refueling temperature. Thus the final justified cooldown limitations in terms of actual temperature are as follows:

RCS Temp., °F	CD Rate, °F/hr
≤ 70	30
> 70	100

These limitations are less limiting as compared with the existing limitations as listed in Section 1.

5.6 Recommendations are provided regarding pressure and temperature instruments that should be used by control room operators to monitor reactor coolant pressure and temperature to prevent violation of the P-T limits and meet LTOP requirements. These are found in Section 4.

6 REFERENCES

6.1 Waterford Unit 3 Technical Specifications, Amendment No. 154, November 1999.

6.2 CENP Letter WS-PS-2000-0003, Rev. 000, "Design Document Research in Support of LTOP Cooldown Rate Improvement", F. P. Ferraraccio to R. Gilmore (Entergy), dated 27 March 2000.

6.3 Entergy Operations ESR No.: 2000-5, approved 25 April 2000.

6.4 NRC Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low temperatures", Rev. 1, Nov. 1988. - Attached to Standard Review Plan 5.2.2, "Overpressure Protection", Rev. 2, Nov. 1988.

6.5 CENP Inter-Office Correspondence C-MECH-93-051, Rev. 000, "Appendix G P-T Limits Pressure Correction Factors", R. Paakkonen to C. Stewart, dated 24 August 1993.

6.6 CENP Engineering Report C-MECH-ER-021, Rev. 000, "Development of Reactor Coolant System Pressure Temperature Limits for 20 Effective Full Power Years (Fluence=2.29×10^{19} n/cm^2) and Recommended Surveillance Withdrawal Schedule for Waterford Unit 3", October 1993.

6.7 CENP Calculation C-PEC-185, Rev. 0, "Mass and Energy Addition Pressure Transients in a Solid Water Reactor Coolant System, For LP&L, Waterford Unit 3", issued 25 January 1978.

6.8 CENP Calculation C-PEC-187, Rev. 0, "Determination of Pressure Transients Caused by a RCP Start, with "Hot" Steam Generators, in a Low Temperature Solid Water Reactor Coolant System", issued 2 March 1978.

6.9 CENP Calculation C-PEC-117, Rev. NA, "Sizing of Relief Valves in the Shutdown Cooling Suction Lines for the Louisiana Power and Light Waterford Plant", issued 13 January 1976.

6.10 General Operating Procedure OP-010-003, Rev. 0, "Plant Startup".

6.11 General Operating Procedure OP-010-005, Rev. 0, "Plant Shutdown".

6.12 System Operating Procedure OP-009-005, Rev. 14, "Shutdown Cooling System".

ATTACHMENT A
OTHER DESIGN DOCUMENT CHECKLIST

Other Design Document Checklist

Instructions: The Independent Reviewer is to complete this checklist for each Other Design Document. This Checklist is to be made part of the Quality Record package, although it need not be made a part of or distributed with the document itself. The second section of this checklist lists potential topics which could be relevant for a particular "Other Design Document". If they are applicable, then the relevant section of the Design Analysis Verification Checklist shall be completed and attached to this checklist. (Sections of the Design Analysis Verification Checklist which are not used may be left blank.)

Section 1: To be completed for all "Other Design Documents"	Yes	N/A
Overall Assessment		
1. Are the results/conclusions correct and appropriate for their intended use?	☒	
2. Are all limitations on the results/conclusions documented?	☒	
Documentation Requirements		
1. Is the documentation legible, reproducible and in a form suitable for filing and retrieving as a Quality Record?	☒	
2. Is the document identified by title, document number and date?	☒	
3. For a complete or page change revision, is there a revision history page?	☐	☒
4. Are all pages identified with the document number including revision number?	☒	☐
5. Do all pages have a unique page number?	☒	☐
6. Does the content clearly identify, as applicable:		
a. objective.	☒	☐
b. design inputs (in accordance with QP 3.2).	☒	☐
c. conclusions.	☒	☐
7. Is the verification status of the document indicated?	☒	
8. If an Independent Reviewer is the supervisor or Project Manager, has authorization as an Independent Reviewer been documented?	☐	☒
Assumptions / Contingencies		
1. Are local assumptions documented, justified and verified?	☐	☒
2. Have Internal and External assumptions and contingencies which must be cleared by CENP or the customer been listed on a Contingencies and Assumptions form?	☐	☒
3. Is the Project Manager responsible for clearing the Assumptions / Contingencies identified on the form?	☐	☒

Other Design Document Checklist

Assessment of Significant Design Changes	Yes	N/A
1. Have significant design-related changes that might impact this document been considered?	☐	☒
2. If any such changes have been identified, have they been adequately addressed?	☐	☒
Selection of Design Inputs		
1. Are the design inputs documented?	☒	
2. Are the design inputs correctly selected and traceable to their source?	☒	
3. Are references as direct as possible to the original source or documents containing collection/tabulations of inputs?	☒	
4. Is the reference notation appropriately specific to the information utilized?	☒	
5. Are the bases for selection of all design inputs documented?	☒	☐
6. Is the verification status of design inputs transmitted from customers appropriate and documented?	☐	☒
7. Is the verification status of design inputs transmitted from CENS appropriate and documented?	☐	☒
8. Is the use of customer-controlled sources such as Tech Specs, UFSARs, etc. authorized, and does the authorization specify amendment level, revision number, etc.?	☒	☐
1. a. Is the document accurate and complete and, if applicable, has proper equipment assembly and/or operational sequencing been detailed?	☒	☐
b. If required, has mock-up testing been performed to verify the document's accuracy, completeness and proper assembly or operational sequencing?	☐	☒
References		
1. Are all references listed?	☒	☐
2. Do the reference citations include sufficient information to assure retrievability and unambiguous location of the referenced material?	☒	☐
3. Do the item numbers in the document agree with the item numbers on the reference?	☒	☐
Section 2: Other Potentially Applicable Topic Areas - use appropriate sections of the Design Analysis Verification Checklist (QP 3.4, Exhibit 3.4 - 3) and attach.	Yes	N/A
Use of Computer Software	☐	☒
Applicable Codes and Standards	☒	☐
Literature Searches and Background Data	☐	☒
Methods	☒	☐
Hand Calculations	☒	☐
List of Computer Software	☐	☒
List of optical disks (CD-ROM), computer disks or Microfiche	☐	☒

Other Design Document Checklist			
From Design Analysis Verification Checklist (QP 3.4, Exhibit 3.4 - 3)			
		IR	
Analytical Techniques (Methods)		Yes	N/A
1.	Are the analytical techniques (methods) described in sufficient detail to judge their appropriateness?	☒	No
2.	Are the analytical techniques used or their application governed by an NRC issued SER?	☐	☒
	If yes, have the applicable SERs been documented?	☐	
	If yes, has the basis for concluding the analysis is in conformance been documented?	☐	
3.	Have analytical techniques incorporated by reference to generic, lead plant or previous cycle analyses been previously verified?	☒	☐
4.	Are any modifications or departures from previously approved analytical techniques or Conventional or Automated Procedures documented and justified?	☐	☒
5.	If superseded approved analytical techniques or engineering procedures are used, is their use justified and approved?	☐	☒
6.	Does the issue date of referenced approved Conventional or Automated Procedures predate their use in this analysis?	☒	☐
Other Elements			
1.	Has a comparison of the results with those of a previous cycle or similar analysis been documented and significant differences explained?	☒	☐
2.	Have applicable Codes (e.g., ASME Code) and standards been appropriately referenced and applied?	☒	☐
3.	Is the information from relevant literature searches/background data adequately documented and referenced?	☐	☒
4.	Are hand calculations correct and appropriately documented?	☒	☐
5.	Is all applicable computer output and input included?	☐	☒
6.	Is all computer software used identified by name and revision identification?	☐	☒

Other Design Document Checklist

Independent Reviewer's Comments				
Comment Number	**Reviewer's Comment**	**Response Required?**	**Author's Response**	**Response Accepted?**
1.	In addition to the comments listed below, editorial comments were forwarded to the author in the form of marked text.	No		NA
2.	All mathematical operations were verified by a hand calculator	No		NA
3.	Item 8 of the checklist section on design inputs asks if the use of the Technical Specifications is specifically authorized and if so, does the authorization specify the amendment level. This was answered in the affirmative as the ESR (Reference 6.3) specifically requests re-evaluation of the TS related to cooldown rates. The amendment level is not specified, however, it is clearly identified in the Reference section.	No		NA
4.	Tables 1, 2, 3, 4. Many of the pressure values are identified in the source reference as interpolated values. This notation should be carried over to this document.	Yes	Corrected	Yes
5.	Table 1. At an RCS temperature of 80.0°F and a heatup rate of 30°F/hr, the pressure should be 732.7 psia.	Yes	Corrected	Yes
6.	Table 2 & Table 4. There are two rows with an RCS termperature value of 105.5°F. The second of these rows should be 105.6°F.	Yes	Corrected	Yes
7.	Table 4. At an RCS temperature of 210°F, the pressure in all four columns should be 2305.7 psia.	Yes	Corrected	Yes
8.	Section 4.1 Most of the references are caution statements. Caution statements typically are placed before the applicable step. For example, in Reference 6.10, the caution statement after 9.2.7.1 is applicable to step 9.2.8. It is more appropriate to reference the step after the caution statement.	Yes	Corrected	Yes
9.	Section 4.1, In the section "From Reference 6.10:," the last bullet should read "... are > 272°F."	Yes	Corrected	Yes
10.	Section 3.2, Minimum Pressure, cites	Yes	Corrected	Yes

	Reference 6.1, Technical Specifications. Should this be Reference 6.6?			
11.	Suggest adding a section titled "Qualilty Assurance," that identifies the quality class and governing procedure (QPM-101). The introduction section contains a statement that this report is safety related, however, a stand-alone section is recommended.	Yes	Added a statement to Section 1.	Yes

Checklist completed by:			
Independent Reviewer:	Roger M. Orsulak	[signature]	6/30/2000
	Printed Name	Signature	Date

ATTACHMENT 4

To W3F1-2001-0032

NPF-38-230

MARK-UP OF AFFECTED TECHNICAL SPECIFICATION BASES PAGES
(Information Only)

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500 °F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

As used in this specification, the term 'cold leg temperature' is intended to be representative of that entering the reactor vessel beltline. During periods with the reactor coolant pumps in operation, the T_{COLD} temperature indication meets this intent. However, during periods when the reactor coolant pumps are not in service, the T_{COLD} temperature indicator is in a stagnant segment of piping and the indication may not necessarily be indicative of that entering the reactor vessel beltline. During the condition when the reactor coolant pumps are operating, the lowest T_{COLD} of a loop with an operating reactor coolant pump is used to monitor the P-T limits. However, during periods when the shutdown cooling system is in operation and following coastdown of the last RCP, the shutdown cooling temperature is the 'cold leg temperature' used to monitor P-T limits.

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60 °F per hour or cooldown rate of up to 100 °F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper and nickel content of the material in question, can be predicted using FSAR Table 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10 CFR Part 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in FSAR Table 5.3-10. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.