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BOUNDLESS ENERGY

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June 15, 2021

AEP-NRC-2021-28
10 CFR 50.90

Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Donald C. Cook Nuclear Plant Unit 2
License Amendment Request Regarding a Change to the Reactor Coolant System Pressure and Temperature Limits and Low Temperature Overpressure Protection (LTOP) System

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 2, is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 2. The proposed amendment will revise the Reactor Coolant System (RCS) heatup and cooldown curves and Low Temperature Overpressure Protection (LTOP) requirements in TS 3.4.3 and 3.4.12, respectively. The proposed changes to the LTOP requirements in TS 3.4.12 will also require changes to be made to TS 3.4.6, 3.4.7, and 3.4.10.

This application for amendment to the CNP Unit 2 TS proposes to revise TS 3.4.3, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits", to update Figures 3.4.3-1 "Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY and during vacuum fill)" and 3.4.3-2 "Reactor Coolant System Pressure versus Temperature Limits – Various Cooldown Rates Limits (Applicable for service period up to 32 EFPY and during vacuum fill)" with revised P/T limits applicable up to 48 Effective Full Power Years (EFPY). A similar request was made for CNP Unit 1 TS, with the subsequent amendment issued January 12, 2021, (ADAMS Accession Number ML20329A001).

In addition, I&M proposes to change CNP Unit 2 TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," to align with an updated LTOP analysis. The proposed changes to the LTOP requirements in TS 3.4.12 will also require changes to be made to TS 3.4.6, 3.4.7, and 3.4.10.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 is an evaluation of the proposed change to Section 3.4.3, 3.4.6, 3.4.7, 3.4.10, and 3.4.12 of the Unit 2 TS. Enclosure 3 contains marked up copies of the applicable Unit 2 TS pages. New Unit 2 TS pages, with proposed changes incorporated, will be provided to the Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested. Enclosure 4 contains marked up copies of the applicable Unit 2 TS Bases pages, provided for information purposes. Changes to the existing TS Bases, consistent with the technical and regulatory analyses, will be implemented under TS 5.5.12 "Technical Specifications (TS) Bases Control Program."

PROPRIETARY INFORMATION
Enclosure 6 to this letter contains proprietary information.
Withhold from public disclosure under 10 CFR 2.390.
Upon removal of Enclosure 6, this Letter is decontrolled.

Enclosure 5 contains WCAP-18456-NP, Revision 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," Westinghouse Electric Company (Non-Proprietary), February 2020. This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P/T) limit curves for normal operation of the CNP Unit 2 reactor vessel.

Enclosure 6 contains LTR-SCS-20-18-P, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Proprietary). This letter transmits the proprietary version of the LTOP analysis report for CNP Unit 2.

Enclosure 7 contains LTR-SCS-20-18-NP, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Non-Proprietary). This letter transmits the non-proprietary version of the LTOP analysis report for CNP Unit 2.

Enclosure 8 contains an affidavit from the Westinghouse Electric Company for withholding the proprietary information contained in Enclosure 6. This affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

Approval of the proposed amendment is requested commensurate with the NRC's normal review schedule of approximately one year, but no later than July 29, 2022. This will allow sufficient time to incorporate these changes into the CNP Unit 2 TS prior to the Unit 2 reactor vessel reaching 32 EFPY, which is currently expected to occur in November of 2022. The license amendment will be implemented within 90 days of NRC approval.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan state officials.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Director, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Q. Shane Lies
Site Vice President

JMT/mll

PROPRIETARY INFORMATION
Enclosure 6 to this letter contains proprietary information.
Withhold from public disclosure under 10 CFR 2.390.
Upon removal of Enclosure 6, this Letter is decontrolled.

Enclosures:

1. Affirmation
2. Evaluation of Proposed Amendment to Revise Unit 2 Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits and Low Temperature Overpressure Protection (LTOP) System for Donald C. Cook Nuclear Plant Unit 2
3. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked To Show Proposed Changes
4. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked To Show Proposed Changes (For Information Only)
5. WCAP-18456-NP, Revision 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," Westinghouse Electric Company (Non Proprietary), February 2020.
6. LTR-SCS-20-18-P, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Proprietary)
7. LTR-SCS-20-18-NP, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Non-Proprietary)
8. Affidavit of Withholding Pursuant to 10 CFR 2.390, Westinghouse Electric Company

c: R. J. Ancona – MPSC
EGLE – RMD/RPS
J.B. Giessner –NRC Region, III
NRC Resident Inspector
R.M. Sistevaris –AEP Ft. Wayne, w/o enclosures
J. E. Walcutt - AEP Ft. Wayne, w/o enclosures
S. P. Wall –NRC Washington, D.C.
A. J. Williamson –AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2021-28

AFFIRMATION

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

Indiana Michigan Power Company

Quinton S. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15th DAY OF June, 2021

Notary Public

My Commission Expires 02/20/2025

Enclosure 2 to AEP-NRC-2021-28

Evaluation of Proposed Amendment to Revise Unit 2 Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits and Low Temperature Overpressure Protection (LTOP) System for Donald C. Cook Nuclear Plant Unit 2

Table of Contents

1.0 SUMMARY DESCRIPTION

Indiana Michigan Power Company (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Unit 2, requests an amendment to the CNP Unit 2 Operating License DPR-74 by incorporating the proposed change for the CNP Unit 2 Technical Specifications (TS). The proposed change is a request to revise TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits" and TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System" for CNP Unit 2. The proposed changes to the LTOP requirements in TS 3.4.12 will also require changes to be made to TS 3.4.6, 3.4.7, and 3.4.10. These changes are necessary to account for a service life increase from 32 Effective Full Power Years (EFPY) to an extended service life of 48 EFPY.

Approval of the proposed amendment is requested commensurate with the NRC's normal review schedule of approximately one year, but no later than July 29, 2022. This will allow sufficient time to incorporate these changes into the CNP Unit 2 TS prior to the Unit 2 reactor vessel reaching 32 EFPY, which is currently expected to occur in November of 2022. The license amendment will be implemented within 90 days of NRC approval.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

The CNP Unit 2 Reactor Coolant System (RCS) consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator (SG). The system also includes a pressurizer, connecting piping, pressurizer safety and relief valves, and relief tank, necessary for operational control.

During operation, the reactor coolant pumps (RCP) circulate pressurized water through the reactor vessel and the four reactor coolant loops. The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. During transient operation, the system's heat capacity attenuates thermal transients generated by the core or SGs.

By appropriate selection of the inertia of the RCPs, the thermal-hydraulic effects are reduced to a safe level during the pump coast down, which would result from a loss-of-flow situation. The layout of the system assures natural circulation capability following a loss-of-flow to permit decay heat removal without overheating the core. Part of the system's piping serves as part of the emergency core cooling system to deliver cooling water to the core during a loss of coolant accident.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray, to minimize pressure variations due to contraction and expansion of the coolant. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank (PRT), where the discharged steam is condensed and cooled by mixing with water.

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 P/T limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. TS 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 Appendix G requirements during the LTOP MODES.

The current LTOP System for pressure relief consists of two power operated relief valves (PORVs), with reduced lift settings, one PORV and one residual heat removal (RHR) suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability. When all RCS cold leg temperatures are ≥ 140°F and two charging pumps are capable of injecting into the RCS, the LTOP System for pressure relief includes all three RCS relief valves (two PORVs and the RHR suction relief valve). Three RCS relief valves are required for redundancy, since one PORV and one RHR suction relief valve have adequate relieving capability to prevent overpressurization at this coolant input capability.

2.2 Current Technical Specifications Requirements

The CNP Unit 2 LCO 3.4.3 "RCS Pressure and Temperature (P/T) Limits" states:

"LCO 3.4.3 *RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in Figures 3.4.3-1 and 3.4.3-2 with:*

a. A maximum heatup of 60°F in any one hour period;

b. A maximum cooldown of 100°F in any one hour period; and

c. A maximum temperature change of ≤ 5°F in any one hour period, during hydrostatic testing operations above system design pressure."

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of non-ductile failure in accordance with 10 CFR 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or inservice leak and hydrostatic (ISLH) testing, their Applicability is at all times in keeping with the concern for non-ductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other TS provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

<u>The CNP Unit 2 LCO 3.4.6 "RCS Loops – Mode 4" states:</u>

"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 removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

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

2. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ 152°F unless the pressurizer water level is < 62% or the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

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In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

<u>The CNP Unit 2 LCO 3.4.7 "RCS Loops – Mode 5, Loops Filled" states:</u>

"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 above the lower tap of the SG wide range level instrumentation by ≥ 418.77 inches.

--NOTES--

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; 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. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ 152°F unless the pressurizer water level is < 62% or the secondary water temperature of each steam generator is < 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.

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In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be above the lower tap of the SG wide range water level instrumentation by ≥ 418.77 inches.

<u>The CNP Unit 2 LCO 3.4.10 "Pressurizer Safety Valves" states:</u>

"LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2411 psig and ≤ 2559 psig."

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.

<u>The CNP Unit 2 LCO 3.4.12 "Low Temperature Overpressure Protection (LTOP) System" states:</u>

"LCO 3.4.12 An LTOP System shall be OPERABLE with one of the following:

A. No safety injection (SI) pump and a maximum of one charging pump capable of injecting into the RCS, except two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations, and the following:

1. The accumulators isolated, except an accumulator may be unisolated when the accumulator is depressurized and vented; and

2. One of the following pressure relief capabilities:

a. Two power operated relief valves (PORVs) with lift settings ≤ 435 psig;

b. *One PORV with a lift setting ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig; or*

c. *The RCS depressurized and an RCS vent of ≥ 2.0 square inches or any single PORV blocked open.*

OR

B. *No SI pump and both charging pumps capable of injecting into the RCS, and the following:*

1. *The accumulators isolated, except an accumulator may be unisolated when the accumulator is depressurized and vented;*

2. *Two PORVs with lift settings ≤ 435 psig;*

3. *The RHR suction relief valve with a setpoint ≤ 450 psig; and*

4. *All RCS cold leg temperatures ≥ 140°F.*

---*NOTE*---
Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ 152°F unless the pressurizer water level is < 62% or the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.
---"

This LCO provides RCS overpressure protection by having a minimum coolant input capability, limiting reactor coolant pump (RCP) startup transients, and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection (SI) pumps and all but one charging pump incapable of injection into the RCS and isolation of the accumulators. RCPs shall not be started when RCS cold leg temperature is ≤ 152°F unless certain requirements are met. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that is available to terminate an increasing pressure event. When all RCS cold leg temperatures are ≥ 140°F, the coolant input capability is allowed to be increased by allowing both charging pumps to be capable of injecting into the RCS. This is acceptable since requiring three RCS relief valves provides adequate pressure relief capacity under these conditions (one of the two PORVs and the RHR suction relief valve are the overpressure protection devices that are available to terminate an increasing pressure event).

2.3 Reason for the Proposed Change

Background

This License Amendment request (LAR) proposes to revise the RCS Heatup, and Cooldown curves; and the LTOP requirements, in order to allow for an increased service life. The current

TS for these curves expire at a service life of 32 EFPY, which is estimated to occur in November of 2022. Enclosure 5 contains calculations which have been performed to establish pressure versus temperature limits for all curves in TS 3.4.3 for a service life extending up to 48 EFPY, which is the accumulated burnup estimated to occur in Fall of 2040, beyond the expiration of the Unit 2 renewed license.

As expected, the revised curves are more restrictive in some operating regions than the existing ones, due to the effects of increased neutron fluence over the life of the reactor vessel, and the associated increase in RT_{NDT} at the ¼ thickness (1/4T) and ¾ thickness (3/4T) locations. Although the revised curves are more restrictive in some operating regions, the current technical specifications are conservative for today's operation and will be from now until the amendment is approved. This would include TS 3.4.6 Note 2 and the 152°F limit on RCP operation. The new curves were developed using the standard Westinghouse methodologies which have been previously reviewed and approved by the NRC for other licensees.

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS P/T conditions.

The existing Unit 2 P/T limits curves required by 10 CFR 50, Appendix G and contained in TS 3.4.3 are applicable up to 32 EFPY. Enclosure 5 to this letter calculated new P/T limit curves applicable to 48 EFPY. The new P/T curves include a neutron fluence evaluation for the Unit 2 reactor vessel extended beltline region. A new LTOP analysis was performed and documented in Enclosure 6 to this letter to ensure the LTOP system prevents RCS over-pressurization for the postulated heat injection and mass injection transients. The new LTOP analysis ensures the revised P/T limits contained in TS 3.4.3 are not exceeded.

The Unit 2 TS 3.4.12 is changed to reflect the requirements of the new analysis documented in Enclosure 6 to this letter. The proposed changes to LCO 3.4.12 reflect the minimum coolant input capability, limiting RCP startup transient, and pressure relief capacity required by the Enclosure 6 analysis.

The proposed changes to the LTOP requirements in 3.4.12 will also require changes to be made to TS 3.4.6, 3.4.7, and 3.4.10.

2.4 Description of the Proposed Change

In the following mark ups, the deletion of text is shown by striking through the current wording and the addition of text is shown by putting the new text in boxes.

The CNP Unit 2 TS 3.4.3 "RCS Pressure and Temperature (P/T) Limits" will be revised as follows:

Replace the existing TS Figure 3.4.3-1 and Figure 3.4.3-2 with the proposed TS Figure 3.4.3-1 and Figure 3.4.3-2 as shown in Enclosure 3.

This change replaces the CNP Unit 2 RCS P/T curves applicable up to 32 EFPY with curves applicable up to 48 EFPY and reflects the analysis in Enclosure 5 to this letter.

The CNP Unit 2 TS 3.4.6 "RCS Loops – Mode 4" will be revised as follows:

"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 removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

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

2. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~152~~ $\boxed{291}$°F unless ~~the pressurizer water level is < 62% or~~ the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

---"

LCO Note 2 is modified to change the temperature below which RCP operation is restricted based upon delta T between the RCS and steam generators, as stated in Section 5.4 of Enclosure 6 to this letter. This restriction exists to ensure that the first RCP start is within the limits of the LTOP design limiting heat injection transient. The 291°F limit is based on the revised LTOP enable temperature and includes RCS temperature instrument uncertainty. Above the LTOP enable temperature limit of 291°F, LTOP restrictions on starting RCPs do not apply.

The CNP Unit 2 TS 3.4.7 "RCS Loops – Mode 5, Loops Filled" will be revised as follows:

"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 above the lower tap of the SG wide range level instrumentation by ≥ 418.77 inches.

---NOTES---

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; 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. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~152~~291°F unless the ~~pressurizer water level is < 62% or the~~ secondary water temperature of each steam generator is < 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.

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LCO Note 3 is modified to change the temperature below which RCP operation is restricted based upon delta T between the RCS and steam generators, as stated in Section 5.4 of Enclosure 6 to this letter. This restriction exists to ensure that the first RCP start is within the limits of the LTOP design limiting heat injection transient. The 291°F limit is based on the revised LTOP enable temperature and includes RCS temperature instrument uncertainty. Above the LTOP enable temperature limit of 291°F, LTOP restrictions on starting RCPs do not apply.

The CNP Unit 2 TS 3.4.10 "Pressurizer Safety Valves" will be revised as follows:

"LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2411 psig and ≤ 2559 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > ~~299~~291°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 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperatures ≤ ~~299~~291°F.	6 hours 24 hours

"

1. The Applicability in Mode 4 was changed to require pressurizer safety valves to be OPERABLE above 291°F. The 291°F limit is based on the revised LTOP enable temperature and includes RCS temperature instrument uncertainty, as stated in Enclosure 6 to this letter. With RCS cold leg temperature ≤291°F, TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System, provides RCS overpressure protection.
2. Condition B.2 was changed to reflect the new LTOP enable temperature of 291°F. Below this temperature TS 3.4.10 does not apply.

The CNP Unit 2 TS 3.4.12 "Low Temperature Overpressure Protection (LTOP) System" will be revised as follows:

"LCO 3.4.12 *An LTOP System shall be OPERABLE with ~~one of~~ the following:*

A. No safety injection (SI) pump ~~and a maximum of one charging pump~~ capable of injecting into the RCS~~, except two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations,~~ and ~~the following~~:

1. The accumulators isolated, except an accumulator may be unisolated when the accumulator ~~is depressurized and vented;~~ pressure is less than the maximum RCS pressure for the existing

RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3;

2. *One of the following pressure relief capabilities:*

   a. ~~*Two power operated relief valves (PORVs) with lift settings ≤ 435 psig*~~ *The residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig and RCS cold leg temperature ≤ 150°F;*

   b. ~~*One PORV with a lift setting ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig; or*~~ *The residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig and at least one RCP running;*

   c. *Two PORVs with lift settings ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig;*

   d. *Two PORVs with lift settings ≤ 435 psig and RCS cold leg temperature ≥ 200°F; or*

   ~~*c*~~*e. The RCS depressurized and an RCS vent of ≥ 2.0 square inches or any single PORV blocked open.*

~~*OR*~~

~~*B. No SI pump and both charging pumps capable of injecting into the RCS, and the following:*~~

~~*1. The accumulators isolated, except an accumulator may be unisolated when the accumulator is depressurized and vented;*~~

~~*2. Two PORVs with lift settings ≤ 435 psig;*~~

~~*3. The RHR suction relief valve with a setpoint ≤ 450 psig; and*~~

~~*4. All RCS cold leg temperatures ≥ 140°F.*~~

---NOTE---
Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~*152*~~*291°F unless the* ~~*pressurizer water level is < 62% or the*~~ *secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.*

APPLICABILITY: *MODE 4 when any RCS cold leg temperature is ≤* ~~*299*~~*291°F,*
MODE 5,
MODE 6 when the reactor vessel head is on."

"ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable when entering MODE 4.

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CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SI pumps capable of injecting into the RCS.	*A.1 Initiate action to verify all SI pumps are not capable of injecting into the RCS.*	*Immediately*
~~*B. Two charging pumps capable of injecting into the RCS, when only one is allowed to be capable of injecting into the RCS.*~~	~~*B.1 Initiate action to verify a maximum of one charging pump is capable of injecting into the RCS.*~~	~~*Immediately*~~
~~C~~B. An accumulator not isolated when the accumulator ~~is not depressurized and vented.~~ pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed by TS 3.4.3.	*~~C~~B.1 Isolate affected accumulator.*	*1 hour*
~~D~~C. Required Action and associated Completion Time of Condition ~~C~~B not met.	*~~D~~C.1 Increase RCS cold leg temperature to > ~~299~~291°F.* *<u>OR</u>* *~~D~~C.2 Depressurize affected accumulator ~~and vent~~ to less than the maximum RCS pressure for existing cold leg temperature allowed in TS 3.4.3.*	*12 hours* *12 hours*

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E~~D. One required RCS relief valve inoperable in MODE 4 while complying with LCO A.2.c or A.2.d.	~~E~~D.1 Restore required RCS relief valve to OPERABLE status.	7 days
~~F~~E. One required RCS relief valve inoperable in MODE 5 or 6 while complying with LCO A.2.c or A.2.d.	~~F~~E.1 Restore required RCS relief valve to OPERABLE status.	24 hours
F. Required RCP not running.	F.1 Do not start a RCP. AND F.2 Enter Condition G.	Immediately Immediately
G. Two or more required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, ~~B~~C, D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ 2.0 square inches or block open a single PORV.	12 hours

TS 3.4.12 is changed to ensure the new LTOP analysis (Enclosure 6) requirements are reflected in the LCO. The previous LTOP analysis, and TS, reflects the requirement to limit RCS mass injection capability to either one or two centrifugal charging pumps (CCP), dependent on RCS temperature and available relief capacity. The new LTOP analysis demonstrates that RCS overpressure protection is provided when the limiting mass injection transient is from two operating charging pumps for the full range of LTOP applicability. Therefore, the restriction on CCPs that may be in operation has been eliminated. Note that the LTOP TS continues to require the safety injection (SI) pumps to be incapable of injecting into the RCS for the full range of LTOP applicability.

The current LTOP TS states that accumulators must be isolated unless depressurized and vented. The proposed LTOP TS states that accumulators must be isolated unless accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3.

The proposed LCO 3.4.12 is structured as a series of five LCO conditions based on relief capabilities, RCS temperature limitations, and RCP status as applicable, that must be met to ensure RCS overpressure protection. Only one of the five LCO conditions must be met to meet the requirements of the LCO. The proposed LCO conditions are described below:

1. The new LTOP analysis demonstrates that the RHR suction safety can accommodate the most limiting mass injection transient for the full range of LTOP applicability, and the most limiting heat inject transient, startup of the first RCP, for RCS temperatures ≤ 150°F. Proposed LCO A.2.a reflects this required relief capability.

2. The new LTOP analysis documents that if a RCP is running then the most limiting heat injection transient cannot occur, and the remaining non-limiting heat injection transients can be accommodated by the RHR suction safety. In addition, the RHR suction safety can accommodate the most limiting mass injection transient for the full range of LTOP applicability. Therefore, the RHR suction safety can provide overpressure protection for the full range of LTOP applicability with one RCP running. Note that the most limiting heat injection transient is the start of the first RCP with temperature asymmetry between the SGs and the RCS, and the non-limiting heat injection transients are inadvertent pressurizer heater operation and loss of decay heat removal. Proposed LCO A.2.b reflects this required relief capability and RCP status.

3. The new LTOP analysis demonstrates that the RHR suction safety and one pressurizer PORV can accommodate the most limiting mass injection and heat injection transients for the full range of LTOP applicability. Two pressurizer PORVs must be OPERABLE for single failure considerations. Proposed LCO A.2.c reflects this required relief capability.

4. The new LTOP analysis demonstrates that one pressurizer PORV can accommodate the most limiting mass injection and heat injection transients if RCS temperature is ≥ 200°F. Two pressurizer PORVs must be OPERABLE for single failure considerations. Proposed LCO A.2.d reflects this required relief capability.

5. The new LTOP analysis demonstrates that a depressurized RCS with an RCS vent of ≥ 2.0 square inches or any single PORV blocked open can accommodate the most limiting

mass injection transient. Note that since a RCP cannot be intentionally started with the RCS vented, the most limiting heat injection transient is not expected to occur. Proposed LCO A.2.e reflects this required relief capability.

Other proposed changes to Unit 2 TS LCO 3.4.12 are as follows:

- The LCO 3.4.12 mode of applicability is changed to MODE 4 when any RCS cold leg temperature is ≤ 291°F.
- The LCO 3.4.12 note for RCP start was changed to add the new LTOP enable temperature (291°F) and to delete the allowance to start RCPs if pressurizer level is < 62%.
- Condition B is deleted. This condition provided actions if two CCPs were capable of injecting into the RCS when only one was allowed.
- Condition C is relabeled Condition B and is reworded as follows:

  > "An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing cold leg temperature allowed by TS 3.4.3."

- Condition D is relabeled Condition C. Action C.1 is reworded to reflect the new LTOP enable temperature (291°F) and C.2 is reworded to reflect the new wording of Condition B.
- Condition E is relabeled as Condition D and is reworded as follows:

  > "One required RCS relief valve inoperable in MODE 4 while complying with LCO A.2.c or A.2.d."

- Condition F is relabeled as Condition E and is reworded as follows:

  > "One required RCS relief valve inoperable in MODE 5 or 6 while complying with LCO A.2.c or A.2.d."

- A new Condition F was added to provide actions if the required RCP was not running. The prescribed actions are to not start a RCP and to enter Condition G immediately.
- The second "OR" statement in Condition G was modified to reflect the new Condition B.

The following three TS Surveillance Requirements will be impacted by the proposed change as shown below.

SURVEILLANCE		FREQUENCY
SR 3.4.12.2	~~Verify no more than the maximum allowed number of charging pumps are capable of injecting into the RCS.~~ Verify the required RCP is running.	In accordance with the Surveillance Frequency Control Program

The current SR 3.4.12.2 was deleted. A new SR 3.4.12.2 to verify that the required RCP was running was added.

SR 3.4.12.3	------------------------------NOTE------------------------------ Valve position may be verified by use of administrative means. --- Verify each accumulator that is required to be isolated is isolated.	In accordance with the Surveillance Frequency Control Program

Clarification wording was added to SR 3.4.12.3, as an accumulator is not always required to be isolated.

SURVEILLANCE		FREQUENCY
SR 3.4.12.8	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ ~~299~~291°F. --- Perform a COT on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program

The LTOP enable temperature was changed to 291°F in the SR 3.4.12.8.

3.0 TECHNICAL EVALUATION

The basis for the proposed changes to the CNP Unit 2 TS RCS P/T Limit Curves is provided in Enclosure 5 to this letter, as described below. In addition, the basis for the proposed changes to the CNP U2 LTOP analysis is provided in Enclosure 6 to this letter, as described below.

Enclosure 5 to this letter contains WCAP-18456-NP, Revision 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," Westinghouse Electric Company, February 2020. (Non-Proprietary). The RCS P/T limit curves were generated using the K_{IC} methodology detailed in the 1998 Edition through the 2000 Addenda of the ASME Code, Section XI, Appendix G. This P/T limit curve generation methodology is consistent with the U.S. Nuclear Regulatory Commission (NRC) approved methodology documented in WCAP-14040-A, Revision 4 (Reference 1). The heatup and cooldown P/T limit curves utilize the Adjusted Reference Temperature (ART) values for CNP Unit 2 calculated using Regulatory Guide 1.99, Revision 2 (Reference 2).

Enclosure 6 to this letter contains LTR-SCS-20-18-P, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Proprietary). The LTOP Power Operated Relief Valves (PORV) setpoints are selected in accordance with NRC approved methodology (Reference 1) such that the peak pressure during the design basis Mass Injection (MI) and Heat Injection (HI) transients will not exceed the isothermal Appendix G P/T limits.

3.1 Evaluation of Neutron Fluence Methodology

The neutron fluence analysis behind the current 32 EFPY P/T limits documented in WCAP-15047 (ML022110334) utilized the DORT discrete ordinates code Version 3.1 (WCAP-13515 Revision 1, provided to the NRC in ML022100438). The updated neutron fluence analysis provided in Enclosure 5 utilizes RAPTOR-M3G and FERRET, which is consistent with the NRC-approved methodology described in WCAP-18124-NP-A. This methodology was used to address both the beltline and extended beltline regions. NOTE: The NRC Safety Evaluation provided in WCAP-18124-NP-A is limited to the traditional RPV beltline region as there is currently no NRC-approved methodology to address the extended beltline region.

In 2014, I&M submitted a LAR, by letter dated April 9, 2014 (ML14101A367), to revise the P/T limits to account for vacuum refill. The NRC issued a request for additional information, by email dated July 21, 2014 (ML14217A325), which required I&M to address the non-beltline region of the current 32 EFPY P/T Limit curves. This request for additional information was addressed by I&M in letters dated August 15, 2014 (ML14230A677), and September 25, 2014 (ML14273A258), and accepted by the NRC in letter dated October 1, 2014 (ML14259A549).

The updated neutron fluence analysis evaluates the beltline and extended beltline regions to generate P/T limits up to 48 EFPY. In line with the conclusions previously provided to the NRC to address the extended beltline up to 32 EFPY, the updated P/T limits analysis provided in Enclosure 5 states that the beltline region continues to be limiting.

Both the current and the updated neutron fluence analyses utilize data from the most recent Surveillance Capsule withdrawal at CNP Unit 2 (WCAP-13515-NP, Revision 1). Typically, P/T limits are updated after removing and analyzing a surveillance capsule, which allows the calculated data to be validated by the capsule data. However, the updated neutron fluence analysis does not rely on updated surveillance capsule data. By letter dated July 31, 2005 (ML052230442), I&M is obligated by NRC Regulatory Commitment, which in summary is as follows:

> I&M will pull and test one additional standby capsule for each unit between 32 EFPY and 48 EFPY to address the peak fluence expected at 60 years. A fluence update will be performed at approximately 32 EFPY when Capsules W (Unit 1) and S (Unit 2) are pulled and tested. A subsequent fluence update will be performed when the standby capsules are pulled and tested between 32 EFPY and 48 EFPY.

This LAR does not change the CNP surveillance capsule withdrawal schedule, and subsequent surveillance capsule analyses will be used to validate the updated neutron fluence values and P/T limits as described in the above NRC Regulatory Commitment.

3.2 Evaluation of the Allowance to have Both CCPs Capable of Injecting Into the RCS

The LTOP analysis contained in Enclosure 6 to this letter states that the design basis MI flowrate is due to both centrifugal charging pumps injecting into the RCS (with letdown isolated) for the full LTOP temperature range. The analysis results demonstrate that with the relief capabilities required by the LTOP analysis, the TS RCS over-pressurization will not occur. That is, the P/T limits of TS 3.4.3 will not be exceeded. Therefore, the LTOP TS allows both CCPs to be capable of injecting into the RCS at all times within the TS applicability.

3.3 Evaluation of the Change in Accumulator Pressure Requirements

The accumulators must be isolated unless accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3. This is a change from the current TS requirement that the accumulators must be isolated unless depressurized and vented. An accumulator that is depressurized to less than the maximum pressure allowed by the P/T limit curves cannot cause RCS over-pressurization. Depressurizing the accumulator to RCS pressure instead of fully depressurizing the accumulator would save the time and effort of fully depressurizing and subsequently pressurizing the accumulator. Therefore, the proposed LTOP TS allows an accumulator to be unisolated in this circumstance. Note that this more closely aligns the DC Cook Unit 2 LTOP TS with the NUREG 1431 Revision 4, Standard Technical Specifications Westinghouse Plants, verbiage for the LTOP LCO (3.4.12).

CNP operational procedures would be changed to ensure that proper controls were in place to support the proposed change.

3.4 Evaluation of the Change in LTOP Relief Capability Requirements

As determined in the LTOP analysis in Enclosure 6 to this letter, the RHR relief valve is a passive component and is not subject to single active failures. In accordance with the Enclosure 6 LTOP analysis, the following RCS relief capabilities must be operable:

- For 60 ≤ T_{RCS} ≤ 150°F with zero through four RCPs running:
  - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the mass injection (MI) and heat injection (HI) transients.

- For 150 < T_{RCS} < 200°F:
  - With zero RCPs running:
    - The RHR suction relief valve, with a setpoint of ≤ 450 psig, is required to be operable and will protect against the MI transient; and
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against the HI transient.

  - With at least one RCP running:
    - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the MI and HI transients.

- For 200 ≤ T_{RCS} ≤ 291°F:
  - With zero RCPs running:
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against both the MI and HI transients.

  - With at least one RCP running:
    - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the MI and HI transients; or
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against both the MI and HI transients.

3.5 Evaluation of the Change for Unit 2 TS 3.4.12 LCO

The proposed LTOP TS requires one of the following relief capabilities to be operable:

1. The residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig and RCS cold leg temperature ≤ 150°F.

   Basis: Proposed LCO A.2.a reflects the equipment availability required by the LTOP analysis contained in Enclosure 6. Per Enclosure 6, the RHR suction safety is capable of providing protection for both the LTOP mass injection and heat injection transients if RCS cold leg temperature is ≤ 150°F. Note the 150°F limit includes RCS temperature instrument uncertainty.

2. The RHR suction relief valve with a setpoint ≤ 450 psig and at least one RCP running.

   Basis: Proposed LCO A.2.b reflects the equipment availability required by the LTOP analysis contained in Enclosure 6. Per Enclosure 6, the RHR suction safety is capable of providing protection for the LTOP mass injection transient for the full range of LTOP applicability. Since the most limiting heat injection transient is the start of the first RCP, the requirement to verify that a RCP is already running ensures that the most limiting heat injection transient cannot occur. Note that Enclosure 6 performed an analysis to ensure that the RHR suction safety alone can prevent RCS over-pressurization during the non-limiting heat injection transients, i.e. inadvertent actuation of pressurizer heaters and loss of RHR cooling.

3. Two PORVs with lift settings ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig.

   Basis: Proposed LCO A.2.c reflects the equipment availability required by the LTOP analysis contained in Enclosure 6. Per Enclosure 6, the RHR suction safety and a single Pressurizer PORV are capable of providing protection for both the LTOP mass injection and heat injection transients for the full range of LTOP applicability. Since Pressurizer PORVs are active components, both PORVs are required to be operable to provide over pressure protection in the event of a failure of one PORV.

4. Two PORVs with lift settings ≤ 435 psig and RCS cold leg temperature ≥ 200°F.

   Basis: Proposed LCO A.2.d reflects the equipment availability required by the LTOP analysis contained in Enclosure 6. Per Enclosure 6, a single Pressurizer PORV is capable of providing protection for both the LTOP mass injection and heat injection transients if RCS cold leg temperature is ≥ 200°F. Since Pressurizer PORVs are active components, both PORVs are required to be operable to provide over pressure protection in the event of a failure of one PORV. Note the 200°F limit includes RCS temperature instrument uncertainty.

5. The RCS depressurized and an RCS vent of ≥ 2.0 square inches or any single PORV blocked open.

   Basis: Proposed LCO A.2.e reflects the equipment availability required by the LTOP analysis contained in Enclosure 6. Per Enclosure 6, the RCS depressurized with an RCS vent of ≥ 2.0 square inches or any single PORV blocked open provides RCS over pressure protection for the full range of LTOP applicability for the mass injection transient. Note this is not a change to the existing LTOP TS requirement, this discussion is included here to confirm that the new analysis contained in Enclosure 6 demonstrated the acceptability of this relief capability.

3.6 Evaluation of the Change for Unit 2 TS 3.4.12 Conditions

- Existing Condition B is deleted in its entirety because the analysis performed in Enclosure 6 allows both charging pumps to be in service for the full range of LTOP applicability.
- Existing Condition C is relabeled as Condition B and reworded to reflect the new requirements for accumulator isolation.
- Existing Condition D is relabeled as Condition C. Action C.1 is modified for the new LTOP enable temperature. Action C.2 is reworded to reflect the new requirements for accumulator isolation. That is, that an accumulator does not need to be isolated if accumulator pressure is less than the P/T limits curve.
- Existing Condition E is relabeled as Condition D. Condition D is modified to only apply when using LCO A.2.c or A.2.d. These LCOs require multiple relief paths operable and it is appropriate to allow time to restore a redundant relief flow path in these cases since an operable relief path remains available. LCO A.2.a and A.2.b require only the RHR suction safety operable, and the appropriate Condition to enter is Condition G if the RHR suction safety is inoperable in these circumstances.
- Existing Condition F is relabeled as Condition E. Condition E is modified to only apply when using LCO A.2.c or A.2.d. These LCOs require multiple relief paths operable and it is appropriate to allow time to restore a redundant relief flow path in these cases since an operable relief path remains available. LCO A.2.a and A.2.b require only the RHR suction safety operable, and the appropriate Condition to enter is Condition G if the RHR suction safety is inoperable in these circumstances.
- A new Condition F was added to provide the actions necessary to take if the RCP required to be running by LCO A.2.b is not running. Action F.1 ensures that a RCP is not started because this could initiate a heat injection transient, and Action F.2 directs entry into Condition G to restore compliance with LTOP pressure relief requirements.
- Condition G was modified to change the second "OR" statement. Failure to comply with the action requirements of Condition B requires entry into Condition C and not Condition G. Therefore, Condition B was removed from the second "OR" statement. This change reflects the renumbering of the LCO Conditions.

3.7 Evaluation of the Change for Unit 2 TS 3.4.12 Surveillances

- The existing SR 3.4.12.2 is deleted. This SR verified no more than the maximum allowed number of charging pumps are capable of injecting into the RCS. The new LTOP analysis allows both charging pumps to be capable of injecting into the RCS at all times. Therefore, this SR is no longer applicable.
- A new SR 3.4.12.2 was added to verify the required RCP is running. If LCO A.2.b is being used to comply with LTOP requirements then one RCP must be running. One RCP running ensures that the design basis limiting heat injection transient cannot occur. This SR periodically verifies the RCP required by LCO A.2.b is running. The specified frequency is in accordance with the surveillance frequency control program.
- Clarification wording was added to SR 3.4.12.3, as an accumulator is not always required to be isolated.
- The note to SR 3.4.12.8 was modified to reflect the new LTOP enable temperature of 291°F.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Regulatory Requirements

The proposed changes were developed in accordance with the following NRC regulations and guidance:

- 10 CFR 50 Appendix G
- Regulatory Guide (RG) 1.99, Radiation Embrittlement of Reactor Vessel Materials, Rev. 2
- ASME B&PV Code Section XI Appendix G, 1998 Edition through the 2000 Addenda
- NRC Regulatory Issue Summary (RIS) 2014-11, Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components, October 14, 2014

10 CFR 50 Appendix G, by reference to ASME B&PV Code Section XI Appendix G specifies fracture toughness and testing requirements for the RCS carbon and low alloy steel materials. 10 CFR 50 Appendix G also requires prediction of the effects of neutron irradiation on vessel embrittlement by calculating the Adjusted Reference Temperature (ART) and the Charpy Upper Shelf Energy (USE). The methods provided in RG 1.99 Rev. 2 (Reference 2), defines the ART as the sum of unirradiated reference temperature, the increase of reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

As described in the CNP Updated Final Safety Analysis Report, Section 1.4, the Plant Specific Design Criteria (PSDC) define the principal criteria and safety objectives for the CNP design. The following PSDC are relevant to the proposed amendment:

PSDC CRITERION 33 Reactor Coolant Pressure Boundary Capability

> *The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.*

The proposed changes are consistent with the above regulatory requirements and criteria. Therefore, the proposed changes will assure safe operation by continuing to meet applicable regulations and requirements.

4.2 Precedent

The methodology under which the heatup and cooldown curves were created is a standard used by Westinghouse throughout the industry. The P/T limit curve generation methodology is consistent with the NRC approved methodology documented in WCAP-14040-A, Revision 4, and has been previously approved by the NRC as listed below.

1. Letter from Scott P. Wall, NRC, to Senior Vice President and Chief Nuclear Officer (Indiana Michigan Power Company, Inc.), "Donald C. Cook Nuclear Plant, Unit No. 1 - Issuance of Amendment No. 356 Re: Updating The Reactor Coolant System Pressure-Temperature Limits (EPID L-2020-LLA-0081)," dated January 12, 2021, (ADAMS Accession Number ML20329A001).

2. Letter from Thomas J. Wengert, NRC, to ANO Site Vice President (Entergy Operations, Inc.), "Arkansas Nuclear One, Unit 2 - Issuance of Amendment Re: Updating the Reactor Coolant System Pressure-Temperature Limits (EPID L-2017-LLA-0396)," dated November 27, 2018, (ADAMS Accession Number ML18298A012).

3. Letter from Douglas V. Pickett, NRC, to Vice President, Operations (Entergy Nuclear Operations, Inc.), "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Changes to Reactor Vessel Heatup and Cooldown Curves and Low Temperature Overpressure Protection system Requirements (TAC No. MF5746)," dated September 3, 2015, (ADAMS Accession Number ML15226A159).

4.3 No Significant Hazards Consideration

This LAR to the CNP Unit 2 TS proposes to revise TS 3.4.3, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits", to update Figures 3.4.3-1 "Reactor Coolant System Pressure versus Temperature Limits – Heatup Limit, Criticality Limit, and Leak Test Limit (Applicable for service period up to 32 EFPY and during vacuum fill)" and 3.4.3-2 "Reactor Coolant System Pressure versus Temperature Limits – Various Cooldown Rates Limits (Applicable for service period up to 32 EFPY and during vacuum fill)" with revised P/T limits applicable up to 48 Effective Full Power Years (EFPY).

In addition, I&M proposes to change CNP Unit 2 TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," to align with an updated LTOP analysis. The proposed changes to the LTOP requirements in 3.4.12 will also require RCS temperature limit changes to be made to TS 3.4.6, 3.4.7, and 3.4.10.

TS Figures 3.4.3-1 and 3.4.3-2 provide the RCS pressure versus temperature limits for various modes of reactor operation. These curves specify safe zones of reactor operation under varying RCS pressure and temperature conditions.

As required by 10 CFR 50.91(a), the CNP analysis of the issue of no significant hazards consideration is presented below:

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

Response: No.

The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves or the LTOP analysis. The proposed changes do not modify the RCS pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected.

Therefore, it is concluded that the proposed amendment does not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. *Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?*

Response: No.

The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the heatup and cooldown limitation curves and LTOP analysis do not affect any activities or equipment other than the RCS pressure boundary and do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

The proposed TS changes do not involve a significant reduction in the margin of safety. The proposed RCS P/T limit curves will continue to provide adequate margins of protection for the reactor coolant pressure boundary (RCPB). The methodologies used in the supporting analyses are in accordance with the criteria set forth in the applicable regulations and do not involve a significant reduction in the margin of safety. The operating limits established by the updated P/T limit curves provide margin against non-ductile failure of the RCPB per the requirements of 10 CFR 50, Appendix G.

Therefore, the proposed amendment does not involve a significant reduction in margin of safety.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

I&M has evaluated the proposed amendments for environmental considerations. The review has resulted in the determination that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

6.0 REFERENCES

1. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

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

Enclosure 3 to AEP-NRC-2021-28

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages
Marked To Show Proposed Changes

3.4.3-3
3.4.3-4
3.4.6-1
3.4.7-1
3.4.10-1
3.4.12-1
3.4.12-2
3.4.12-3
3.4.12-4
3.4.12-5

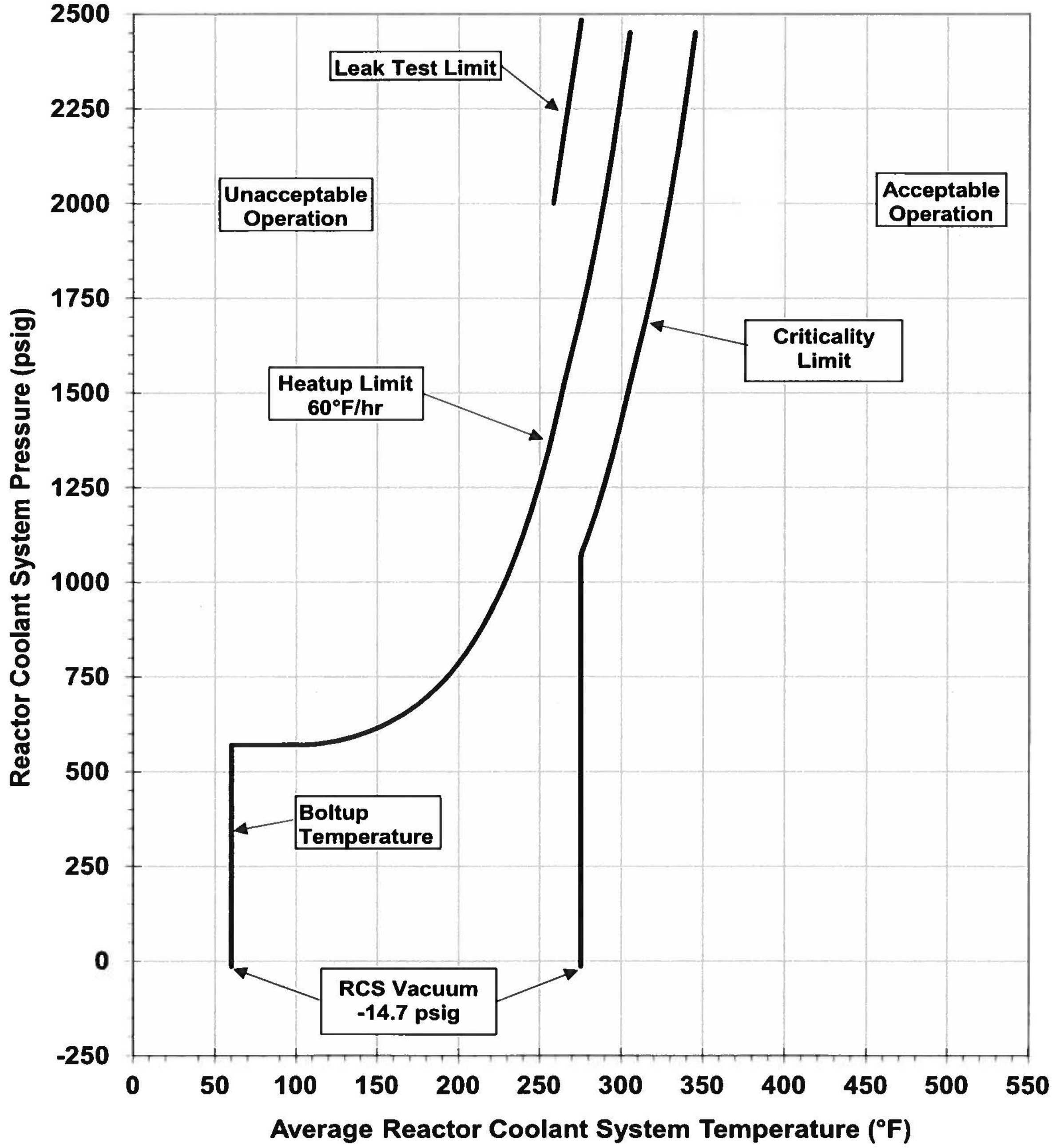

Figure 3.4.3-1 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits -
Heatup Limit, Criticality Limit, and Leak Test Limit
(Applicable for service period up to ~~32~~48 EFPY and during vacuum fill)

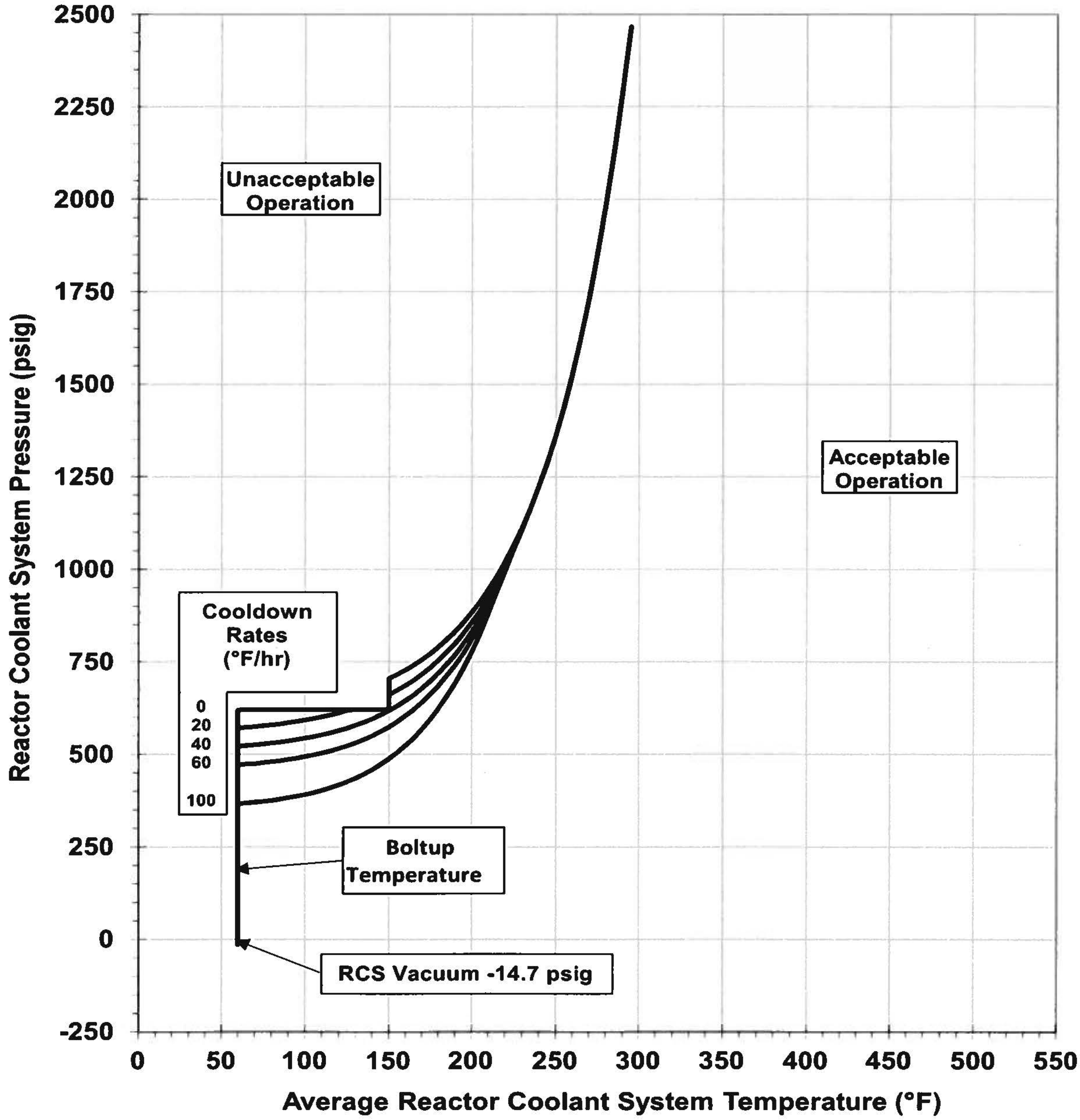

Figure 3.4.3-2 (page 1 of 1)
Reactor Coolant System Pressure versus Temperature Limits - Various Cooldown Rates Limits
(Applicable for service period up to ~~32~~48 EFPY and during vacuum fill)

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 removed from operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; and

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

2. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~152~~ $\boxed{291}$°F unless ~~the pressurizer water level is < 62% or~~ the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required loop inoperable.	A.1	Initiate action to restore a second loop to OPERABLE status.	Immediately
	<u>AND</u>		
	A.2	--------------NOTE-------------- Only required if RHR loop is OPERABLE. ------------------------------------- Be in MODE 5.	24 hours

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 above the lower tap of the SG wide range level instrumentation by ≥ 418.77 inches.

---NOTES---

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"; 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. Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~152~~291°F unless the ~~pressurizer water level is < 62% or the~~ secondary water temperature of each steam generator is < 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 ≥ 2411 psig and ≤ 2559 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > ~~299~~ 291°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 ≤ ~~299~~ 291°F.	6 hours 24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with one of the following:

A. No safety injection (SI) pump ~~and a maximum of one charging pump~~ capable of injecting into the RCS~~, except two charging pumps may be made capable of injecting into the RCS for ≤ 1 hour for pump swap operations,~~ and ~~the following~~:

1. The accumulators isolated, except an accumulator may be unisolated when the accumulator ~~is depressurized and vented;~~ pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3;

2. One of the following pressure relief capabilities:

a. ~~Two power operated relief valves (PORVs) with lift settings ≤ 435 psig~~ The residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig and RCS cold leg temperature ≤ 150°F;

b. ~~One PORV with a lift setting ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig; or~~ The residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig and at least one RCP running;

c. Two PORVs with lift settings ≤ 435 psig and the residual heat removal (RHR) suction relief valve with a setpoint ≤ 450 psig;

d. Two PORVs with lift settings ≤ 435 psig and RCS cold leg temperature ≥ 200°F; or

~~c~~e. The RCS depressurized and an RCS vent of ≥ 2.0 square inches or any single PORV blocked open.

~~OR~~

~~B. No SI pump and both charging pumps capable of injecting into the RCS, and the following:~~

~~1. The accumulators isolated, except an accumulator may be unisolated when the accumulator is depressurized and vented;~~

~~2. Two PORVs with lift settings ≤ 435 psig.~~
~~3. The RHR suction relief valve with a setpoint ≤ 450 psig; and~~

~~4. All RCS cold leg temperatures ≥ 140°F.~~

---NOTE---
Reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ ~~152~~291°F unless the ~~pressurizer water level is < 62% or the~~ secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~299~~291°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable when entering MODE 4.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SI pumps capable of injecting into the RCS.	A.1 Initiate action to verify all SI pumps are not capable of injecting into the RCS.	Immediately
~~B. Two charging pumps capable of injecting into the RCS, when only one is allowed to be capable of injecting into the RCS.~~	~~B.1 Initiate action to verify a maximum of one charging pump is capable of injecting into the RCS.~~	~~Immediately~~
~~C~~B. An accumulator not isolated when the accumulator ~~is not depressurized and vented.~~ pressure is greater than or equal to the maximum RCS pressure for the existing cold leg	~~C~~B.1 Isolate affected accumulator.	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
temperature allowed by TS 3.4.3.		
~~D~~C. Required Action and associated Completion Time of Condition ~~C~~B not met.	~~D~~C.1 Increase RCS cold leg temperature to > ~~299~~291°F.	12 hours
	OR	
	~~D~~C.2 Depressurize affected accumulator ~~and vent~~ to less than the maximum RCS pressure for existing cold leg temperature allowed in TS 3.4.3.	12 hours
~~E~~D. One required RCS relief valve inoperable in MODE 4 while complying with LCO A.2.c or A.2.d.	~~E~~D.1 Restore required RCS relief valve to OPERABLE status.	7 days
~~F~~E. One required RCS relief valve inoperable in MODE 5 or 6 while complying with LCO A.2.c or A.2.d.	~~F~~E.1 Restore required RCS relief valve to OPERABLE status.	24 hours
F. Required RCP not running.	F.1 Do not start a RCP.	Immediately
	AND	
	F.2 Enter Condition G.	Immediately
G. Two or more required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, ~~B~~C, D, E, or F not met. OR	G.1 Depressurize RCS and establish RCS vent of ≥ 2.0 square inches or block open a single PORV.	12 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify no SI pumps are capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	~~Verify no more than the maximum allowed number of charging pumps are capable of injecting into the RCS.~~ Verify the required RCP is running.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	------------------------------NOTE------------------------------ Valve position may be verified by use of administrative means. -- Verify each accumulator that is required to be isolated is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	Verify RHR suction isolation valves are open for the required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.5	Verify required RCS vent ≥ 2.0 square inches open or a single PORV blocked open.	In accordance with the Surveillance

SURVEILLANCE		FREQUENCY
		Frequency Control Program
SR 3.4.12.6	Verify PORV block valve is open for each required PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.7	Verify pressure in each required emergency air tank bank is ≥ 900 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.8	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ ~~299~~291°F. -- Perform a COT on each required PORV, excluding actuation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.9	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	In accordance with the Surveillance Frequency Control Program

Enclosure 4 to AEP-NRC-2021-28

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked To Show Proposed Changes (For Information Only)

B 3.4.3-1 to B 3.4.3-6
B 3.4.6-2
B 3.4.7-2
B 3.4.10-1
B 3.4.10-3
B 3.4.12-1 to B 3.4.12-16

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This LCO contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, criticality, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. Vacuum fill of the RCS is performed in Mode 5 under sub-atmospheric pressure and isothermal RCS conditions. Vacuum fill is an acceptable condition since the resulting pressure/ temperature combination is reflected on the operating limits provided in Figures 3.4.3-1 and 3.4.3-2. Insert 1

The LCO establishes operating limits that provide a margin to ~~brittle~~non-ductile failure of the ~~reactor vessel and piping of the~~ reactor coolant pressure boundary (RCPB)~~. The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.~~ Insert 2.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to ~~brittle~~non-ductile failure during normal operation, anticipated operational occurrences, and system inservice hydrostatic leak tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section ~~III~~ XI, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically using the methodology provided in Regulatory Guide 1.99, Revision 2. These calculated values are periodically confirmed by

BASES

BACKGROUND (continued)

removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings ~~and the recommendations of Regulatory Guide 1.99 (Ref. 6)~~ using the methodology provided in Appendix G to the ASME Section XI Code (Ref 2).

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 2 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~~ challenge the margins against non-ductile 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. 7), 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. ~~Reference 1 establishes the methodology for determining the P/T limits~~. 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(c)(2)(ii).

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. Because the pressurizer is subjected to insurges and outsurges and it is used to control RCS pressure, it experiences higher heatup and cooldown rates which have been analyzed separately. ~~These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure~~.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the ~~stress~~ analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed ~~to provide guidance~~ for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for non-ductile failure~~. The limits do not apply to the pressurizer~~.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," LCO 3.4.2, "RCS Minimum Temperature for Criticality," and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for ~~pressure and~~ temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for non-ductile failure, and ~~stress~~ analyses

BASES

APPLICABILITY (continued)

have been performed for normal maneuvering profiles, such as power ascension or descent.

BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by ~~stress~~ analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed within 72 hours. The evaluation must include an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the RCS. Several methods may be used, including comparison with pre-analyzed ~~transients~~ conditions in the ~~stress~~ analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If any Required Action and associated Completion Time of Condition A is not met, the unit must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of

BASES

ACTIONS (continued)

time or a sufficiently severe event resulted in a determination that the RCS is or may be unacceptable for continued operation. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation ~~with~~of undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the unit to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. The evaluation must include an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the RCS. Several methods may be used, including comparison with pre-analyzed ~~transients~~conditions in the ~~stress~~ analyses, or inspection of the components.

BASES

ACTIONS (continued)

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within limits is required when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~WCAP-15047, Rev. 2, dated May 2002~~. WCAP-18456-NP, Rev. 0, dated February 2020

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section ~~III~~XI, Appendix G.

4. ASTM E 185-82, July 1982.

5. 10 CFR 50, Appendix H.

6. Regulatory Guide 1.99, Revision 2, May 1988.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

T.S. Bases 3.4.3 Insert 1

Operation is permitted in the region located to the right and below the curves provided in Figures 3.4.3-1 and 3.4.3-2. Conversely, operation in the region located to the left and above the curves is not permitted. These curves were developed without allowance for instrumentation uncertainties. The curves in the plant operating procedures are adjusted to account for the instrumentation uncertainties associated with the actual instruments used to implement these curves.

T.S. Bases 3.4.3 Insert 2

components fabricated from low alloy steel. The reactor vessel is the most limiting RCPB component subjected to neutron irradiation embrittlement. However, the remainder of the RCPB components fabricated from low alloy steel (e.g., steam generators, pressurizer, etc.) have also been considered in the analysis. These components were analyzed to the applicable ASME Code Section III Editions and met the requirements at the time of construction.

BASES

LCO (continued)

Utilization of the Note is permitted provided the following conditions are met:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained 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 ~~or the pressurizer water level be < 62%~~ before the start of an RCP with any RCS cold leg temperature ≤ ~~152~~291°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, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump (either the east or west) 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. Management of gas voids is important to RHR System OPERABILITY.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

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 water level above the lower tap of the SG wide range level instrumentation by ≥ 418.77 inches. 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 water levels above the lower tap of the SG wide range level instrumentation by ≥ 418.77 inches. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit the RHR pump to be removed from operation when switching operation from one RHR loop or flowpath to another. The 1 hour time period is adequate to switch the RHR loops, 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:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet the requirements of LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained 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~~ or the pressurizer water level be < 62%~~ before the start of a~~n~~ reactor coolant pump (RCP) with an RCS cold leg temperature < ~~152~~291°F. This restriction

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 Trip 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 complete loss of steam flow to the turbine. 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. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ ~~299~~ $\boxed{291}$°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 and lower pressure limits are based on the ± 3% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. 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 Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABILITY (continued)

The LCO is not applicable in MODE 4 when any RCS cold leg temperatures are ≤ ~~299~~$\boxed{291}$°F or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with 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.

B.1 and B.2

If Required Action A.1 and associated Completion Time is not met or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ ~~299~~$\boxed{291}$°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. With any RCS cold leg temperatures at or below ~~299~~$\boxed{291}$°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.

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. ITS 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 P/T limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability, limiting reactor coolant pump (RCP) startup transients, and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection (SI) pumps ~~and all but one charging pump~~ incapable of injection into the RCS and isolation of the accumulators. RCPs shall not be started when RCS cold leg temperature is ≤ ~~152~~291°F unless certain requirements are met. The pressure relief capacity requires adequate capacity available ~~either two redundant RCS relief valves~~ or a depressurized RCS and an RCS vent of sufficient size. ~~One~~ Sufficient RCS relief capacity ~~valve~~ or the open RCS vent is the overpressure protection ~~device~~ that is available to terminate an increasing pressure event. ~~When all RCS cold leg temperatures are ≥ 140°F, the coolant input capability is allowed to be increased by allowing both charging pumps to be capable of injecting into the RCS. This is acceptable since requiring three RCS relief valves provides adequate pressure relief capacity under these conditions (one of the two PORVs and the RHR suction relief valve are the overpressure protection devices that are available to terminate an increasing pressure event).~~

BASES

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not specifically require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core 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 or~~ an SI pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of one of the following: ~~two power operated relief valves (PORVs), with reduced lift settings, one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability. When all RCS cold leg temperatures are ≥ 140°F and two charging pumps are capable of injecting into the RCS, the LTOP System for pressure relief includes all three RCS relief valves (two PORVs and the RHR suction relief valve). Three RCS relief valves are required for redundancy, since one PORV and one RHR suction relief valve have adequate relieving capability to prevent overpressurization at this coolant input capability.~~

1. The RHR suction relief valve with RCS temperature ≤ 150°F;
2. The RHR suction relief valve with one RCP running;
3. Two power operated relief valves (PORVs), with reduced lift settings, and the RHR suction relief valve;
4. Two power operated relief valves (PORVs), with reduced lift settings, with RCS temperature ≥ 200°F; or
5. The RCS depressurized and an RCS vent of ≥ 2.0 square inches or any single PORV blocked open.

Note that the temperatures used above include allowances for RCS temperature instrument uncertainties.

PORV Requirements

When the RCS temperature is below the LTOP enable temperature, a safeguards circuit can be ~~is~~ manually armed which allows the PORVs to open in the event of a low temperature overpressurization transient. RCS pressure is monitored by two wide range pressure instruments with each instrument providing an opening signal to one PORV.

BASES

BACKGROUND (continued)

The LTOP setpoints for both PORVs are the same. Having the setpoints of both valves within the limit 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.

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 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 valve is exposed to the RCS and is able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valve OPERABLE for RCS overpressure mitigation. The RHR suction relief valve is a 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. 3) for Class 2 relief valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain ~~the~~RCS pressure within limits ~~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.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, blocking open any one of the three PORVs, and disabling its block valve in the open position, or similarly establishing a vent by opening sufficient RCS vent valves to provide a 2.0 square inch vent path. 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. 4) 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 ~~299~~291°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At ~~299~~291°F and below, overpressure prevention is provided by one of the RCS relief paths required by this LCO. ~~falls to two OPERABLE RCS relief valves (or three RCS relief valves when all RCS cold leg temperatures are ≥ 140°F and two charging pumps are capable of injecting into the RCS) or to a depressurized RCS and a sufficient sized RCS vent.~~. 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 limit curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met ~~using the RCS relief valve method or the depressurized and vented RCS condition~~.

The LCO contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 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:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

<u>Heat Input Type Transients</u>

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~~ that the LTOP overpressure protection means cannot handle:

a. Rendering all SI pumps ~~and all but one charging pump~~ incapable of injection~~, unless all RCS cold leg temperatures are ≥ 140°F, and~~;

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

c. Disallowing a startup of an RCP with one or more RCS cold leg temperatures ≤ 291 ~~152~~°F, unless ~~the pressurizer water level is < 62% or~~ the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

The Reference 4 analyses demonstrate the following:

1. The RHR suction safety can accommodate the most limiting mass injection transient for the full range of LTOP applicability, and the most limiting heat injection transient, startup of the first RCP, for RCS temperatures ≤ 150°F.
2. If a RCP is running then the most limiting heat injection transient cannot occur, and the remaining non-limiting heat injection transients and the limiting mass injection transient can be accommodated by the RHR suction safety. Therefore, the RHR suction safety can provide overpressure protection for the full range of LTOP applicability with one or more RCPs running.
3. The RHR suction safety and one pressurizer PORV can accommodate the most limiting mass injection and heat injection transients for the full range of LTOP applicability. Two pressurizer PORVs must be OPERABLE for single failure considerations.
4. One pressurizer PORV can accommodate the most limiting mass injection and heat injection transients if RCS temperature is ≥ 200°F. Two pressurizer PORVs must be OPERABLE for single failure considerations.
5. A depressurized RCS with an RCS vent of ≥ 2.0 square inches or any single PORV blocked open can accommodate the most limiting mass injection and heat injection transients. Note that since a RCP cannot be intentionally started with the RCS vented, the most limiting heat injection transient is not expected to occur.

~~either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one charging pump is actuated. Thus, the LCO allows only one charging pump to be capable of injecting into the RCS during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent~~ The LTOP analysis does not analyze ~~handle the pressure transient need from~~ accumulator injection ~~, when RCS temperature is low,~~. Therefore, the LCO also requires ~~the~~ accumulators isolation when the accumulators are not depressurized to below the P/T limits curve for the given RCS temperature. ~~and vented.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The analyses also demonstrate that one PORV and one RHR suction relief valve can maintain RCS pressure below limits when both charging pumps are actuated, all RCS cold leg temperatures are ≥ 140°F. Thus, the LCO allows two charging pumps to be capable of injecting into the RCS under these conditions.~~

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at ≤ ~~299~~291°F. This value includes RCS temperature instrument uncertainty.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the specified setpoint. The setpoints are derived by analyses that model the performance of the LTOP System~~, assuming the mass addition transient of one or two charging pumps injecting into the RCS or the limiting heat input transient of an RCP startup with temperature asymmetry within the RCS or between the RCS and steam generators of 50°F above each of the RCS cold leg temperatures~~. A single PORV can provide protection for the most severe heat injection transient for the full range of LTOP applicability, and can provide protection for the most severe mass injection transient if RCS temperature is ≥ 200°F. 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, as necessary, when the P/T limits are revised. 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.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

RHR Suction Relief Valve Performance

Analyses show that the RHR suction relief valve with a setpoint ≤ 450 psig ~~will pass flow greater than that required for the mass addition transient of one charging pump injecting into the RCS while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the mass addition event, The RHR suction relief valve~~ will maintain RCS pressure to within the Appendix G limit curves and 110% of the RHR System design pressure (660 psig) during the most severe mass injection transient, and the most severe heat injection transient if RCS temperature is ≤ 150°F. ~~When all RCS cold leg temperatures are ≥ 140°F and two charging pumps are capable of injecting into the RCS, the RHR suction relief valve and one PORV, in combination, will maintain RCS pressure less than the P/T limit curve.~~

If at least one RCP is running then the most limiting heat injection transient cannot occur. Analysis show that the RHR suction safety is capable of maintaining RCS pressure within the Appendix G limit curves during the non-limiting heat injection transients for the full range of LTOP applicability. Therefore, the RHR suction safety will maintain RCS pressure to within the Appendix G limit curves and 110% of the RHR System design pressure (660 psig) during the most severe mass injection transient, and the applicable heat injection transients for the full LTOP temperature range if at least one RCP is running.

As the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valve must be analyzed to still accommodate the design basis transients for LTOP.

The RHR suction relief valve is a passive component and is not subject to active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.0 square inches or a single blocked open PORV is capable of mitigating the allowed LTOP overpressure transients. The capacity of a vent this size is greater than the flow of the mass addition transient for the LTOP configuration of ~~one~~ two charging pumps 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The LTOP System satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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.

~~To limit the coolant input capability,~~ Tthe LCO restricts coolant injection capability to two charging pumps, i.e. the safety injection pumps must be incapable of injection into the RCS. In addition, all accumulators must be isolated, or depressurized to below the P/T limits curve value for the given RCS temperature. ~~provides two options. The first option requires that no SI pumps and a maximum of one charging pump be capable of injecting into the RCS, and all accumulators isolated (i.e., the discharge isolation valves closed and deactivated).~~

~~The first option, however, allows two charging pumps to be made capable of injecting into the RCS for ≤ 1 hour during pump swap operations. One hour provides sufficient time to safely complete the actual transfer and to complete the administrative controls and Surveillance Requirements associated with the swap. The intent is to minimize the actual time that more than one charging pump is physically capable of injection.~~ In addition, an accumulator may be unisolated when the accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3 ~~is depressurized and vented~~. This permits the accumulator discharge isolation valve Surveillance to be performed only when ~~under these~~ the pressure and temperature limits of the P/T limit curve are not exceeded ~~conditions~~.

Furthermore, the ~~first~~ LCO options requires ~~one of~~ the ~~three~~ following pressure relief capabilities, as applicable:

a. Two OPERABLE PORVs;

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit required by the LCO and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits. Motive power for

BASES

LCO (continued)

the PORVs is through the use of air. Normally this air is supplied by the plant control air source. To assure OPERABILITY of the PORVs in the event of a loss of control air, a backup air supply is provided. The backup air supply consists of compressed air bottles (the emergency air tank bank), piping, and valves. The backup air supply contains enough air to support PORV operation for 10 minutes with no operator action upon a loss of control air. Only two of the three PORVs have a backup air supply, therefore they are the only PORVs that can be used to meet the LCO requirements.

b. ~~One OPERABLE PORV and one~~ An OPERABLE RHR suction relief valve; ~~or~~

   An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valves are open, its setpoint is ≤ 450 psig, and testing has proven its ability to open at this setpoint.

c. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 2.0 square inches or a single blocked open PORV.

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

~~Consistent with the first option, the second option requires that no SI pumps be capable of injecting into the RCS and that the accumulators are isolated, except an accumulator may be unisolated when it is depressurized and vented. However, the second option allows both charging pumps to be capable of injecting into the RCS, provided all RCS cold leg temperatures are ≥ 140°F and all three of the relief valves (two PORVs and one RHR suction relief valve) described in the first option are OPERABLE~~.

The ~~Both~~ LCO options are modified by a Note that places restrictions on RCP startups. This is necessary to ensure the limiting heat input transient is maintained within the analyses assumptions. Therefore, the Note states that reactor coolant pumps shall not be started with one or more RCS cold leg temperatures ≤ 291 ~~152~~°F unless ~~the pressurizer water level is < 62% or~~ the secondary water temperature of each steam generator is < 50°F above each of the RCS cold leg temperatures.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ ~~299~~ 291°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that

BASES

LCO (continued)

meets the Reference 1 P/T limits with all RCS cold leg temperatures > ~~299~~291°F. When the reactor vessel head is off, overpressurization cannot occur.

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 with all RCS cold leg temperatures > ~~299~~291°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 resulting in little or no time available to allow operator action to mitigate the event.

ACTIONS

A Note prohibits the application of LCO 3.0.4.b to an inoperable LTOP system when entering MODE 4. There is an increased risk associated with entering MODE 4 from MODE 5 with LTOP inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

A.1 ~~and B.1~~

With one or more SI pumps capable of injecting into the RCS, RCS overpressurization is possible. ~~In addition, when only one charging pump is allowed to be capable of injecting into the RCS and both charging pumps are actually capable, 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.

B~~C~~.1, C~~D~~.1, and C~~D~~.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator is not depressurized to less than the maximum RCS pressure for existing cold leg temperature allowed in TS 3.4.3 ~~and vented~~.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C~~D~~.1 and Required Action C~~D~~.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > ~~299~~291°F, an accumulator pressure of 658 psig cannot exceed the LTOP limits if the accumulators are fully injected.

BASES

ACTIONS (continued)

Depressurizing the accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed in TS 3.4.3~~and venting the affected accumulators~~ also gives this protection.

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.

~~E~~D.1

In MODE 4 when any RCS cold leg temperature is ≤ ~~299~~291°F and while complying with LCO A.2.c or A.2.d, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two or three RCS relief valves ~~(depending upon the condition of the charging pumps) in any combination of the PORVs and the RHR suction relief valve~~ are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

This condition can be used only while complying with LCO A.2.c or A.2.d when more than one relief valve is required to be OPERABLE. At least one additional relief valve is OPERABLE. Therefore, it is appropriate to allow some time to restore an inoperable relief valve to operable status.

The Completion Time considers the facts that only one or two of the RCS relief valves (depending upon RCS temperature~~the condition of the charging pumps~~) are required to mitigate an overpressure transient and that the likelihood of a single active failure of the remaining valve path(s) during this time period is very low.

~~F~~E.1

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

This condition can be used only while complying with LCO A.2.c or A.2.d when more than one relief valve is required to be OPERABLE. At least one additional relief valve is OPERABLE. Therefore, it is appropriate to allow some time to restore an inoperable relief valve to operable status.

BASES

ACTIONS (continued)

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 the minimum OPERABLE RCS relief valve(s) required to protect against overpressure events.

F.1 and F.2

If the RCP required by LCO A.2.b is not running the RHR suction safety may not be able to provide overpressure protection for a heat injection transient. RCS flow should not be re-initiated since it could cause a heat injection transient. Since the LTOP system may not be able to provide overpressure protection for the heat injection transient in this condition it is appropriate to immediately enter Condition G.

~~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 the minimum OPERABLE RCS relief valve(s) required to protect against overpressure events.~~

G.1

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

a. Two or more required RCS relief valves are inoperable;

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

The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F (e.g., when an RCP is started without meeting the requirements of the Note to LCO 3.4.12).

In addition, if complying with LCO A.2.a or A.2.b only the RHR suction safety valve is required to be operable. If the RHR suction safety valve becomes inoperable when it is the only RCS relief valve available the appropriate condition to enter is Condition G

The vent must be sized ≥ 2.0 square inches or the vent must be a blocked open PORV 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.

BASES

ACTIONS (continued)

The Completion Time considers the time required to place the unit 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~~, SR 3.4.12.2,~~ and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, no SI pumps ~~and a maximum of one or two charging pumps (depending upon whether the LCO Option A or B is being used)~~ are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and deactivated. The SI pump(s) ~~and charging pump~~ 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 RCS injection such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed, or at least one valve in the discharge flow path being closed and sealed or locked.

In addition, SR 3.4.12.3 is modified by a Note that allows the accumulator discharge isolation valve position to be verified by administrative means. This is acceptable since the valve position was verified prior to deactivating the valve, access to the containment is restricted, and valves are only operated under strict procedural control.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.2

This SR requires verification that the required RCP is in operation and circulating reactor coolant. This surveillance is only required if complying with LCO 3.4.12.A.2.b. Verification includes flow rate, temperature, or pump status monitoring, which help ensure RCS forced flow. The existence of forced flow from at least one RCP ensures that the limiting heat injection transient, startup of the first RCP, cannot occur. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

The required RHR suction relief valve shall be demonstrated OPERABLE by verifying the RHR suction isolation valves are open. 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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.5

The RCS vent of ≥ 2.0 square inches or a blocked open PORV is proven OPERABLE by verifying its open condition. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The passive vent path arrangement must only be open if the vent is being used to satisfy the pressure relief requirements of LCO 3.4.12.A.2.~~c~~e.

SR 3.4.12.6

The PORV block valve must be verified open 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 ~~one or more~~ PORVs are required to satisfy 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.7

Verification that each required emergency air tank bank's pressure is ≥ 900 psig assures adequate air pressure for reliable PORV operation. With the emergency air supply at ≥ 900 psig, there will be enough air to support PORV operation for 10 minutes with no operator action upon a loss of control air. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.8

Performance of a COT is required on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the LCO limit. PORV actuation could depressurize the RCS and is not required.

A Note has been added indicating that this SR is not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ ~~299~~291°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 if PORVs are required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. 10 CFR 50, Appendix G.

2. Generic Letter 88-11.

3. ASME, Boiler and Pressure Vessel Code, Section III.

4. ~~WCAP-13235, "Donald C. Cook Units 1 & 2, Analysis of Low Temperature Overpressurization Mass Injection Events with Pressurizer Steam Bubble and RHR Relief Valve, March 1992; "WCAP-12483 Revision 1, "Analysis of Capsule U From the American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program, December 2002;" and WCAP-13515, Revision 1, "Analysis of Capsule U From Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program, May 2002."~~

Westinghouse Letter LTR-SCS-20-18-NP, D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY, Revision 0.

5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.

Enclosure 5 to AEP-NRC-2021-28

WCAP-18456-NP, Revision 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," Westinghouse Electric Company, February 2020. (Non-Proprietary)

Westinghouse Non-Proprietary Class 3

WCAP-18456-NP
Revision 0

February 2020

D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

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Revision 0

D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the D.C. Cook Unit 2 reactor vessel. The P-T limit curves were generated using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G. This P-T limit curve generation methodology is consistent with the U.S. Nuclear Regulatory Commission (NRC) approved methodology documented in WCAP-14040-A, Revision 4. The heatup and cooldown P-T limit curves utilize the Adjusted Reference Temperature (ART) values for D.C. Cook Unit 2 calculated using Regulatory Guide 1.99, Revision 2. The limiting ART values in material with a postulated axial flaw were those of the Intermediate Shell Plate 10-1 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations.

The P-T limit curves were generated for 48 effective full-power years (EFPY) using a heatup rate of 60°F/hr, and cooldown rates of 0° (steady-state), -20°, -40°, -60°, and -100°F/hr. The curves were developed with the flange requirements of 10 CFR 50, Appendix G, but the curves were developed without margins for instrumentation errors. The curves can be found in Figures 8-1 and 8-2.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 48 EFPY.

Appendix B contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix B, all of the other ferritic RCPB components meet the applicable requirements of Section III of the ASME Code.

Appendix C contains the credibility evaluation of the D.C. Cook Unit 2 reactor vessel surveillance data per the requirements of Regulatory Guide 1.99, Revision 2. D.C. Cook Unit 2 fluence values, described in Section 2.0, were used to complete the evaluation.

Appendix D provides the validation of the radiation transport calculation models based on neutron dosimetry measurement.

1 INTRODUCTION

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) of the beltline region material of the reactor vessel. The adjusted RT_{NDT} is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (T_{NDT}) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} ($RT_{NDT(U)}$). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The NRC has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown P-T limit curves documented in this report were generated using the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [2]. Specifically, the K_{Ic} methodology from Section XI, Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Code [3] was used. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors.

The purpose of this report is to present the calculations and the development of the D.C. Cook Unit 2 heatup and cooldown P-T limit curves for 48 EFPY. This report documents the calculated ART values and the development of the P-T limit curves for normal operation. The calculated ART values for 48 EFPY are documented in Section 7 of this report. The fluence projections used in the calculation of the ART values are provided in Section 2 of this report, and a validation of the radiation transport calculation model based on neutron dosimetry measurements is contained in Appendix D.

The P-T limit curves herein were generated without instrumentation errors. The reactor vessel flange requirements of 10 CFR 50, Appendix G [4] have been incorporated in the P-T limit curves. Discussion of the other RCPB ferritic components relative to P-T limits is contained in Appendix B.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

Discrete ordinates (S_N) transport analyses were performed to determine the neutron radiation environment within the reactor pressure vessel (RPV). In these analyses, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. The dosimetry analysis documented in Appendix D shows that the ±20% (1σ) acceptance criteria specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [5], is met, based on the measurement-to-calculation (M/C) comparison results for the in-vessel surveillance capsules withdrawn and analyzed to-date. Additional information regarding compliance with Regulatory Guide 1.190 is provided in Appendix D. These validated calculations form the basis for providing projections of the neutron exposure of the RPV through the end of license extension (EOLE).

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET" [6]. The neutron transport evaluation methodology described in [6] is based on the guidance of Regulatory Guide 1.190. Note, however, that the NRC Safety Evaluation Report (SER) in [6] states that the applicability of the methodology described in [6] is limited to the traditional RPV beltline region approximated by the RPV region near the active height of the core.

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the RPV, a series of fuel-cycle-specific forward transport calculations were performed using the three-dimensional discrete ordinates code, RAPTOR-M3G [6], and the BUGLE-96 cross-section library [7]. The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for light water reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{12} order of angular quadrature. Energy- and space-dependent core power distributions were treated on a fuel-cycle-specific basis.

The D.C. Cook Unit 2 reactor is a standard Westinghouse 4-loop design employing reactor internals that include 1.125-inch-thick baffle plates and a fully circumferential thermal shield. The model of the reactor (and reactor cavity) geometry used in the plant-specific evaluation is shown in Figure 2-1 through Figure 2-3.

The model extends radially from the center of the core to 349.89 cm, azimuthally from 0° to 45° (taking advantage of the octant symmetry of the reactor configuration), and axially from -380.26 cm to 358.75 cm with respect to the midplane of the active core. Elevations of key RPV materials relative to the model geometry are provided in Table 2-1.

A plan view of the model geometry at the core midplane is shown in Figure 2-1. In this figure, a single octant is depicted showing the arrangement of the core, reactor internals, core barrel, thermal shield, downcomer, cladding, RPV, reactor cavity, reflective insulation, and bioshield. Depictions of the in-vessel surveillance capsules, including their associated support structures, are also shown.

From a neutronics standpoint, the inclusion of the surveillance capsules and associated support structures in the geometric model is significant. Since the presence of the capsules and support structures has a marked impact on the magnitude of the neutron fluence rate and relative neutron and gamma ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are accounted for in the transport calculations.

Section views of the model geometry are shown in Figure 2-2 and Figure 2-3. Note that the stainless steel former plates located between the core baffle and barrel regions are shown in these figures.

When developing the reactor model shown in Figure 2-1 through Figure 2-3, nominal design dimensions were employed for the various structural components. Likewise, water temperatures and, hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids and guide tubes.

The geometric mesh description of the reactor model shown in Figure 2-1 through Figure 2-3 consisted of 220 radial by 188 azimuthal by 374 axial intervals. Mesh sizes were chosen to ensure sufficient resolution of the stair-step-shaped baffle plates as well as an adequate number of meshes throughout the radial and axial regions of interest. The pointwise inner iteration convergence criterion utilized in the calculations was set at a value of 0.001.

The core power distributions used in the plant-specific transport analysis were taken from nuclear design documentation. The data extracted included fuel assembly-specific initial enrichments, beginning-of-cycle burnups and end-of-cycle burnups. Appropriate axial power distributions were also obtained.

For each fuel cycle of operation, fuel-assembly-specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope-dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel-assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the spatial mesh arrays used in the discrete ordinates calculations.

In Table 2-1, axial and azimuthal locations of the RPV materials are provided. The axial position of each material is indexed to $z = 0.0$ cm, which corresponds to the midplane of the active fuel stack.

Cycle-specific calculations were performed for Cycles 1 through 24, with core thermal powers given in Table 2-2. Note that future fluence projection data beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21 through 23, but include a 1.1 bias on the core thermal power. Note that at the time of development of the fluence model, Cycle 24 was yet to be completed and, thus, the results for this cycle are based on the cycle design data, whereas Cycles 21 through 23 had completed and, as such, the results for these cycles are based on actual operating data. Therefore, only Cycles 21 through 23, being the most recently completed cycles, are used for projections per [28].

Neutron fluence rate and fluence for the RPV are given in Table 2-3, Table 2-4, and Table 2-7. Similarly, iron atom displacement rate and iron atom displacements for the RPV are provided in Table 2-5, Table 2-6, and Table 2-8. The data presented represent the maximum neutron exposures experienced by RPV materials. The reported data also consider both the inner and outer radius of the RPV base metal, and account for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for each operating cycle of the reactor. Note that for any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

Calculated neutron exposure projections of the RPV are provided in Table 2-4 and Table 2-6 through Table 2-8. These projections are based on the average spatial power distributions and reactor operating conditions of Cycles 21 through 23, but include a 1.1 bias on the core thermal power. The projected results will remain valid as long as future plant operation is consistent with these assumptions.

Results of the discrete ordinates transport analyses pertinent to the surveillance capsule evaluations are provided in Table 2-9 through Table 2-11. In Table 2-9, the calculated fast neutron fluence rate and fluence ($E > 1.0$ MeV) are provided at the geometric center of the capsules and at core midplane as a function of operating time. Similar data presented in terms of iron atom displacement rate (dpa/s) and integrated iron atom displacements (dpa) are given in Table 2-10.

In Table 2-11, lead factors associated with the surveillance capsules are provided as a function of operating time. The lead factor is defined as the ratio of the neutron fluence ($E > 1.0$ MeV) at the geometric center of the surveillance capsule to the maximum neutron fluence ($E > 1.0$ MeV) at the pressure vessel clad/base metal interface.

All surveillance capsules at the 40° first-octant-equivalent (FOE) azimuthal locations have been removed from the RPV, so neutron exposure data at the 40° FOE azimuthal positions beyond Cycle 8 are unnecessary (because there are no capsules receiving any fluence). However, if any capsules were to be re-inserted or re-located to the 40° FOE azimuthal locations, it would be necessary to know the fast neutron fluence rate at the surveillance capsule holder position(s). To allow for the determination of potential fast fluence accumulation, the projected fast fluence rate ($E > 1.0$ MeV) at each surveillance capsule location is provided in Table 2-12. Projections of future operation are based on the spatial power distributions and reactor operating conditions of Cycles 21 through 23, but include a 1.1 bias on the core thermal power. This bias is intended to account for cycle-to-cycle variations in peripheral fuel assembly relative powers that are expected to occur during the time period of future operation evaluated in this report. Note that RPV neutron exposure rates are dominated by neutron leakage from the peripheral fuel assemblies. The additional fast fluence accumulated for any re-inserted/re-located capsule can be determined by multiplying the fast fluence rate value in Table 2-12 for the appropriate capsule position with the irradiation duration in effective full-power seconds (EFPS).

2.3 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the RPV is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology used in the plant-specific neutron exposure evaluation is carried out in the following four stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator (NUREG/CR-6454, "Pool Critical Assembly Pressure Vessel Facility Benchmark" [8]) at the Oak Ridge National Laboratory (ORNL) and the VENUS-1 experiment.
2. Comparison of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment (NUREG/CR-6453, "H.B. Robinson-2 Pressure Vessel Benchmark" [9]).
3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments (WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" [6]).
4. Comparison of the calculations with all available dosimetry results from the RPV measurement programs carried out at the D.C. Cook Unit 2 (Appendix D).

The first phase of the methods qualification (PCA comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross sections. This phase, however, did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational and geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational method approximations as well as to a lack of knowledge relative to various plant-specific parameters. The overall calculational uncertainty applicable to the D.C. Cook Unit 2 analyses was established from the results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons of plant-specific dosimetry measurements) was used solely to demonstrate the adequacy of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used to bias the final results in any way.

Table 2-13 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in WCAP-18124-NP-A. The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in Appendix D support these uncertainty assessments for D.C. Cook Unit 2.

Table 2-1 RPV Material Locations

Material	Axial Elevation[(a)] (cm)	Azimuth[(b)] (Degrees)
Outlet Nozzle to Upper Shell Weld – Lowest Extent	262.30	22.0
Inlet Nozzle to Upper Shell Weld – Lowest Extent	254.76	23.0
Upper Shell	241.90 to 358.75[(c)]	0.0 to 45.0
Upper Shell to Intermediate Shell Circumferential Weld	236.79 to 241.90	0.0 to 45.0
Intermediate Shell	-32.47 to 236.79	0.0 to 45.0
Intermediate Shell Longitudinal Welds		
Weld 1 (170°)[(d),(e)]	-32.47 to 236.79	9.25 to 10.75
Weld 2 (350°)[(d),(e)]	-32.47 to 236.79	9.25 to 10.75
Intermediate Shell to Lower Shell Circumferential Weld	-39.25 to -32.47	0.0 to 45.0
Lower Shell	-305.83 to -39.25	0.0 to 45.0
Lower Shell Longitudinal Welds		
Weld 1 (90°)[(d),(e)]	-305.83 to -39.25	0.0 to 0.75
Weld 2 (270°)[(d),(e)]	-305.83 to -39.25	0.0 to 0.75
Lower Shell to Lower Head Circumferential Weld	-311.22 to -305.83	0.0 to 45.0

Note(s):

(a) Values listed are indexed to Z = 0.0 at the midplane of the active fuel stack.
(b) Azimuthal angles are given relative to the cardinal axes at 0°, 90°, 180°, and 270°.
(c) Elevation given is equal to the maximum elevation of the reactor model.
(d) Azimuthal angles are given relative to 0° as shown on reactor vessel drawing
(e) This weld is approximately 2.2 inches in width. At the RPV inner radius (86.719 inches), this corresponds to ~1.5°.

Table 2-2 Reactor Core Power Level

Cycle	Core Thermal Power (MWt)
1	3411
2	3411
3	3411
4	3411
5	3411
6	3411
7	3411
8	3411
9	3411
10	3411
11	3411
12	3411
13	3411
14	3468[a]
15	3468
16	3468
17	3468
18	3468
19	3468
20	3468
21	3468
22	3468
23	3468
24	3468

Note(s):

(a) A reactor power uprate from 3411 MWt to 3468 MWt was implemented at the beginning of this cycle.

Table 2-3 Calculated Maximum Fast Neutron Fluence Rate (E > 1.0 MeV) at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence Rate (n/cm^2-s)				
			0°	15°	30°	45°	Maximum[(a)]
1	1.08	1.08	6.63E+09	1.10E+10	1.39E+10	2.16E+10	2.17E+10
2	0.91	1.99	7.41E+09	1.15E+10	1.39E+10	2.21E+10	2.22E+10
3	1.23	3.22	6.98E+09	1.05E+10	1.19E+10	1.78E+10	1.79E+10
4	0.92	4.14	7.04E+09	1.18E+10	1.29E+10	1.84E+10	1.85E+10
5	1.11	5.25	7.14E+09	1.06E+10	1.13E+10	1.64E+10	1.64E+10
6	1.17	6.42	6.09E+09	1.07E+10	1.15E+10	1.73E+10	1.73E+10
7	1.12	7.54	6.72E+09	1.13E+10	1.11E+10	1.60E+10	1.60E+10
8	1.12	8.66	5.50E+09	8.00E+09	1.11E+10	1.59E+10	1.60E+10
9	1.16	9.82	5.12E+09	9.06E+09	1.21E+10	1.59E+10	1.59E+10
10	1.14	10.96	4.27E+09	7.18E+09	1.06E+10	1.43E+10	1.43E+10
11	1.23	12.20	4.23E+09	7.27E+09	1.16E+10	1.78E+10	1.78E+10
12	1.40	13.60	4.06E+09	6.68E+09	8.63E+09	1.26E+10	1.27E+10
13	1.04	14.64	3.99E+09	6.69E+09	8.10E+09	1.17E+10	1.17E+10
14	1.18	15.82	4.85E+09	8.18E+09	1.04E+10	1.59E+10	1.59E+10
15	1.32	17.14	4.42E+09	8.05E+09	1.08E+10	1.66E+10	1.66E+10
16	1.35	18.49	4.69E+09	7.91E+09	9.81E+09	1.37E+10	1.37E+10
17	1.35	19.84	4.25E+09	7.47E+09	1.04E+10	1.66E+10	1.66E+10
18	1.37	21.21	4.92E+09	7.01E+09	8.80E+09	1.32E+10	1.32E+10
19	1.28	22.48	4.83E+09	7.43E+09	9.35E+09	1.41E+10	1.41E+10
20	1.40	23.89	4.43E+09	8.07E+09	1.06E+10	1.74E+10	1.74E+10
21	1.31	25.20	4.47E+09	6.92E+09	9.73E+09	1.54E+10	1.54E+10
22	1.40	26.60	4.34E+09	6.83E+09	9.27E+09	1.42E+10	1.42E+10
23	1.14	27.74	4.02E+09	7.11E+09	1.04E+10	1.53E+10	1.53E+10
24[(b)]	1.37	29.12	4.08E+09	7.46E+09	1.09E+10	1.64E+10	1.65E+10

Note(s):

(a) Values correspond to an azimuthal angle of 44°.

(b) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

Table 2-4 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence (n/cm^2)				
			0°	15°	30°	45°	Maximum[(a)]
1	1.08	1.08	2.26E+17	3.76E+17	4.73E+17	7.35E+17	7.37E+17
2	0.91	1.99	4.38E+17	7.02E+17	8.66E+17	1.36E+18	1.36E+18
3	1.23	3.22	7.08E+17	1.11E+18	1.33E+18	2.05E+18	2.05E+18
4	0.92	4.14	9.14E+17	1.45E+18	1.70E+18	2.59E+18	2.59E+18
5	1.11	5.25	1.16E+18	1.82E+18	2.10E+18	3.16E+18	3.17E+18
6	1.17	6.42	1.39E+18	2.21E+18	2.53E+18	3.80E+18	3.81E+18
7	1.12	7.54	1.62E+18	2.61E+18	2.92E+18	4.36E+18	4.37E+18
8	1.12	8.66	1.82E+18	2.89E+18	3.31E+18	4.92E+18	4.93E+18
9	1.16	9.82	2.01E+18	3.22E+18	3.75E+18	5.50E+18	5.52E+18
10	1.14	10.96	2.16E+18	3.48E+18	4.13E+18	6.02E+18	6.03E+18
11	1.23	12.20	2.32E+18	3.76E+18	4.57E+18	6.70E+18	6.72E+18
12	1.40	13.60	2.50E+18	4.05E+18	4.95E+18	7.25E+18	7.27E+18
13	1.04	14.64	2.64E+18	4.27E+18	5.22E+18	7.63E+18	7.65E+18
14	1.18	15.82	2.82E+18	4.58E+18	5.60E+18	8.22E+18	8.24E+18
15	1.32	17.14	3.00E+18	4.91E+18	6.05E+18	8.90E+18	8.93E+18
16	1.35	18.49	3.20E+18	5.25E+18	6.47E+18	9.47E+18	9.50E+18
17	1.35	19.84	3.38E+18	5.57E+18	6.91E+18	1.02E+19	1.02E+19
18	1.37	21.21	3.59E+18	5.87E+18	7.29E+18	1.07E+19	1.08E+19
19	1.28	22.48	3.79E+18	6.17E+18	7.67E+18	1.13E+19	1.13E+19
20	1.40	23.89	3.98E+18	6.53E+18	8.14E+18	1.21E+19	1.21E+19
21	1.31	25.20	4.17E+18	6.81E+18	8.54E+18	1.27E+19	1.27E+19
22	1.40	26.60	4.36E+18	7.12E+18	8.95E+18	1.33E+19	1.33E+19
23	1.14	27.74	4.51E+18	7.37E+18	9.32E+18	1.39E+19	1.39E+19
24[(b)]	1.37	29.12	4.68E+18	7.69E+18	9.79E+18	1.46E+19	1.46E+19
Future[(c)]	--	36.00	5.71E+18	9.35E+18	1.21E+19	1.81E+19	1.81E+19
Future[(c)]	--	42.00	6.60E+18	1.08E+19	1.42E+19	2.12E+19	2.13E+19
Future[(c)]	--	48.00	7.49E+18	1.22E+19	1.62E+19	2.43E+19	2.44E+19
Future[(c)]	--	54.00	8.38E+18	1.37E+19	1.82E+19	2.74E+19	2.75E+19
Future[(c)]	--	60.00	9.27E+18	1.51E+19	2.03E+19	3.05E+19	3.06E+19

Note(s):

(a) Values correspond to an azimuthal angle of 44°.

(b) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

(c) Values beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-5 Calculated Maximum Iron Atom Displacement Rate at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Displacement Rate (dpa/s)				
			0°	15°	30°	45°	Maximum[(a)]
1	1.08	1.08	1.07E-11	1.77E-11	2.24E-11	3.50E-11	3.51E-11
2	0.91	1.99	1.19E-11	1.85E-11	2.25E-11	3.59E-11	3.59E-11
3	1.23	3.22	1.13E-11	1.68E-11	1.92E-11	2.89E-11	2.90E-11
4	0.92	4.14	1.13E-11	1.88E-11	2.09E-11	2.99E-11	2.99E-11
5	1.11	5.25	1.15E-11	1.69E-11	1.82E-11	2.65E-11	2.66E-11
6	1.17	6.42	9.87E-12	1.70E-11	1.86E-11	2.79E-11	2.80E-11
7	1.12	7.54	1.09E-11	1.80E-11	1.79E-11	2.59E-11	2.59E-11
8	1.12	8.66	8.82E-12	1.28E-11	1.79E-11	2.58E-11	2.59E-11
9	1.16	9.82	8.24E-12	1.45E-11	1.95E-11	2.57E-11	2.58E-11
10	1.14	10.96	6.87E-12	1.15E-11	1.70E-11	2.31E-11	2.32E-11
11	1.23	12.20	6.81E-12	1.17E-11	1.87E-11	2.86E-11	2.87E-11
12	1.40	13.60	6.53E-12	1.07E-11	1.39E-11	2.04E-11	2.04E-11
13	1.04	14.64	6.41E-12	1.07E-11	1.31E-11	1.89E-11	1.89E-11
14	1.18	15.82	7.80E-12	1.31E-11	1.68E-11	2.55E-11	2.56E-11
15	1.32	17.14	7.11E-12	1.29E-11	1.74E-11	2.66E-11	2.67E-11
16	1.35	18.49	7.54E-12	1.27E-11	1.58E-11	2.21E-11	2.22E-11
17	1.35	19.84	6.85E-12	1.20E-11	1.67E-11	2.68E-11	2.69E-11
18	1.37	21.21	7.89E-12	1.12E-11	1.42E-11	2.13E-11	2.14E-11
19	1.28	22.48	7.77E-12	1.19E-11	1.51E-11	2.28E-11	2.28E-11
20	1.40	23.89	7.13E-12	1.29E-11	1.71E-11	2.81E-11	2.82E-11
21	1.31	25.20	7.19E-12	1.11E-11	1.57E-11	2.48E-11	2.49E-11
22	1.40	26.60	6.97E-12	1.09E-11	1.49E-11	2.29E-11	2.29E-11
23	1.14	27.74	6.49E-12	1.14E-11	1.68E-11	2.47E-11	2.48E-11
24[(b)]	1.37	29.12	6.57E-12	1.20E-11	1.76E-11	2.66E-11	2.67E-11

Note(s):

(a) Values correspond to an azimuthal angle of 44°.

(b) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

Table 2-6 Calculated Maximum Iron Atom Displacements at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Displacements (dpa)				
			0°	15°	30°	45°	Maximum[a]
1	1.08	1.08	3.65E-04	6.01E-04	7.63E-04	1.19E-03	1.19E-03
2	0.91	1.99	7.03E-04	1.12E-03	1.40E-03	2.19E-03	2.20E-03
3	1.23	3.22	1.14E-03	1.77E-03	2.14E-03	3.32E-03	3.32E-03
4	0.92	4.14	1.47E-03	2.32E-03	2.75E-03	4.19E-03	4.20E-03
5	1.11	5.25	1.87E-03	2.91E-03	3.39E-03	5.11E-03	5.13E-03
6	1.17	6.42	2.24E-03	3.54E-03	4.07E-03	6.14E-03	6.16E-03
7	1.12	7.54	2.62E-03	4.18E-03	4.70E-03	7.05E-03	7.07E-03
8	1.12	8.66	2.93E-03	4.63E-03	5.33E-03	7.96E-03	7.98E-03
9	1.16	9.82	3.23E-03	5.16E-03	6.05E-03	8.90E-03	8.93E-03
10	1.14	10.96	3.48E-03	5.57E-03	6.66E-03	9.74E-03	9.77E-03
11	1.23	12.20	3.74E-03	6.01E-03	7.37E-03	1.08E-02	1.09E-02
12	1.40	13.60	4.02E-03	6.48E-03	7.98E-03	1.17E-02	1.18E-02
13	1.04	14.64	4.23E-03	6.83E-03	8.40E-03	1.24E-02	1.24E-02
14	1.18	15.82	4.53E-03	7.32E-03	9.02E-03	1.33E-02	1.33E-02
15	1.32	17.14	4.82E-03	7.86E-03	9.75E-03	1.44E-02	1.44E-02
16	1.35	18.49	5.14E-03	8.39E-03	1.04E-02	1.53E-02	1.54E-02
17	1.35	19.84	5.43E-03	8.90E-03	1.11E-02	1.65E-02	1.65E-02
18	1.37	21.21	5.77E-03	9.39E-03	1.17E-02	1.74E-02	1.74E-02
19	1.28	22.48	6.09E-03	9.87E-03	1.23E-02	1.83E-02	1.83E-02
20	1.40	23.89	6.40E-03	1.04E-02	1.31E-02	1.95E-02	1.96E-02
21	1.31	25.20	6.70E-03	1.09E-02	1.38E-02	2.05E-02	2.06E-02
22	1.40	26.60	7.01E-03	1.14E-02	1.44E-02	2.15E-02	2.16E-02
23	1.14	27.74	7.24E-03	1.18E-02	1.50E-02	2.24E-02	2.25E-02
24[b]	1.37	29.12	7.53E-03	1.23E-02	1.58E-02	2.36E-02	2.36E-02
Future[c]	--	36.00	9.17E-03	1.50E-02	1.95E-02	2.93E-02	2.94E-02
Future[c]	--	42.00	1.06E-02	1.73E-02	2.28E-02	3.43E-02	3.44E-02
Future[c]	--	48.00	1.20E-02	1.96E-02	2.61E-02	3.93E-02	3.94E-02
Future[c]	--	54.00	1.35E-02	2.19E-02	2.94E-02	4.44E-02	4.45E-02
Future[c]	--	60.00	1.49E-02	2.42E-02	3.26E-02	4.94E-02	4.95E-02

Note(s):

(a) Values correspond to an azimuthal angle of 44°.

(b) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

(c) Values beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-7 Calculated Maximum Fast Neutron Fluence (E > 1.0 MeV) at the Pressure Vessel Welds and Shells

Material	Fast Neutron Fluence[(a)] (n/cm^2)		
	29.12 EFPY	36 EFPY	42 EFPY
Outlet Nozzle Forging to Upper Shell Welds	1.48E+16	1.87E+16	2.20E+16
Inlet Nozzle Forging to Upper Shell Welds	2.10E+16	2.64E+16	3.11E+16
Upper Shell	7.27E+16	9.21E+16	1.09E+17
Upper Shell to Intermediate Shell Circumferential Weld	1.11E+17	1.40E+17	1.66E+17
Intermediate Shell	1.46E+19	1.81E+19	2.13E+19
Intermediate Shell to Lower Shell Circumferential Weld	1.44E+19	1.78E+19	2.08E+19
Lower Shell	1.46E+19	1.81E+19	2.12E+19
Lower Shell to Lower Vessel Head Circumferential Weld	2.71E+15	3.36E+15	3.93E+15
Intermediate Shell Longitudinal Welds at 10°	6.30E+18	7.66E+18	8.84E+18
Lower Shell Longitudinal Welds at 0°	4.71E+18	5.74E+18	6.64E+18

Material	Fast Neutron Fluence[(a)] (n/cm^2)		
	48 EFPY	54 EFPY	60 EFPY
Outlet Nozzle Forging to Upper Shell Welds	2.54E+16	2.88E+16	3.21E+16
Inlet Nozzle Forging to Upper Shell Welds	3.59E+16	4.06E+16	4.53E+16
Upper Shell	1.26E+17	1.43E+17	1.59E+17
Upper Shell to Intermediate Shell Circumferential Weld	1.91E+17	2.17E+17	2.42E+17
Intermediate Shell	2.44E+19	2.75E+19	3.06E+19
Intermediate Shell to Lower Shell Circumferential Weld	2.39E+19	2.69E+19	2.99E+19
Lower Shell	2.42E+19	2.73E+19	3.04E+19
Lower Shell to Lower Vessel Head Circumferential Weld	4.50E+15	5.07E+15	5.64E+15
Intermediate Shell Longitudinal Welds at 10°	1.00E+19	1.12E+19	1.24E+19
Lower Shell Longitudinal Welds at 0°	7.53E+18	8.43E+18	9.33E+18

Note(s):

(a) Fluence projection for future cycles are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-8 Calculated Maximum Iron Atom Displacements at the Pressure Vessel Welds and Shells

Material	Displacements[a] (dpa)		
	29.12 EFPY	36 EFPY	42 EFPY
Outlet Nozzle Forging to Upper Shell Welds	5.47E-05	6.78E-05	7.93E-05
Inlet Nozzle Forging to Upper Shell Welds	6.88E-05	8.54E-05	9.98E-05
Upper Shell	1.41E-04	1.79E-04	2.11E-04
Upper Shell to Intermediate Shell Circumferential Weld	2.10E-04	2.65E-04	3.13E-04
Intermediate Shell	2.36E-02	2.94E-02	3.44E-02
Intermediate Shell to Lower Shell Circumferential Weld	2.33E-02	2.89E-02	3.38E-02
Lower Shell	2.34E-02	2.91E-02	3.40E-02
Lower Shell to Lower Vessel Head Circumferential Weld	2.04E-05	2.53E-05	2.96E-05
Intermediate Shell Longitudinal Welds at 10°	1.01E-02	1.23E-02	1.42E-02
Lower Shell Longitudinal Welds at 0°	7.56E-03	9.22E-03	1.07E-02

Material	Displacements[a] (dpa)		
	48 EFPY	54 EFPY	60 EFPY
Outlet Nozzle Forging to Upper Shell Welds	9.07E-05	1.02E-04	1.14E-04
Inlet Nozzle Forging to Upper Shell Welds	1.14E-04	1.29E-04	1.43E-04
Upper Shell	2.44E-04	2.76E-04	3.09E-04
Upper Shell to Intermediate Shell Circumferential Weld	3.61E-04	4.10E-04	4.58E-04
Intermediate Shell	3.94E-02	4.45E-02	4.95E-02
Intermediate Shell to Lower Shell Circumferential Weld	3.87E-02	4.36E-02	4.85E-02
Lower Shell	3.90E-02	4.39E-02	4.89E-02
Lower Shell to Lower Vessel Head Circumferential Weld	3.39E-05	3.81E-05	4.24E-05
Intermediate Shell Longitudinal Welds at 10°	1.61E-02	1.80E-02	1.99E-02
Lower Shell Longitudinal Welds at 0°	1.21E-02	1.35E-02	1.50E-02

Note(s):

(a) Fluence projection for future cycles are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-9 Calculated Fast Neutron Fluence Rate and Fluence (E > 1.0 MeV) at the Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence Rate (n/cm^2-s)		Fluence (n/cm^2)	
			4°	40°	4°	40°
1	1.08	1.08	2.09E+10	7.22E+10	7.11E+17	2.46E+18
2	0.91	1.99	2.28E+10	7.23E+10	1.37E+18	4.54E+18
3	1.23	3.22	2.18E+10	5.95E+10	2.22E+18	6.85E+18
4	0.92	4.14	2.18E+10	6.05E+10	2.85E+18	8.61E+18
5	1.11	5.25	2.24E+10	5.41E+10	3.64E+18	1.05E+19
6	1.17	6.42	1.94E+10	5.68E+10	4.35E+18	1.26E+19
7	1.12	7.54	2.14E+10	5.22E+10	5.10E+18	1.44E+19
8	1.12	8.66	1.69E+10	5.28E+10	5.70E+18	1.63E+19
9	1.16	9.82	1.58E+10	5.35E+10	6.28E+18	1.83E+19
10	1.14	10.96	1.30E+10	4.79E+10	6.75E+18	2.00E+19
11	1.23	12.20	1.27E+10	5.75E+10	7.24E+18	2.22E+19
12	1.40	13.60	1.22E+10	4.09E+10	7.79E+18	2.40E+19
13	1.04	14.64	1.21E+10	3.79E+10	8.18E+18	2.53E+19
14	1.18	15.82	1.48E+10	5.13E+10	8.74E+18	2.72E+19
15	1.32	17.14	1.35E+10	5.35E+10	9.30E+18	2.94E+19
16	1.35	18.49	1.44E+10	4.49E+10	9.91E+18	3.13E+19
17	1.35	19.84	1.31E+10	5.37E+10	1.05E+19	3.36E+19
18	1.37	21.21	1.50E+10	4.27E+10	1.11E+19	3.55E+19
19	1.28	22.48	1.49E+10	4.56E+10	1.17E+19	3.73E+19
20	1.40	23.89	1.37E+10	5.64E+10	1.23E+19	3.98E+19
21	1.31	25.20	1.38E+10	4.98E+10	1.29E+19	4.19E+19
22	1.40	26.60	1.33E+10	4.57E+10	1.35E+19	4.39E+19
23	1.14	27.74	1.24E+10	5.02E+10	1.39E+19	4.57E+19
24[(a)]	1.37	29.12	1.26E+10	5.35E+10	1.45E+19	4.80E+19
Future[(b)]	--	36.00	--	--	1.76E+19	5.96E+19
Future[(b)]	--	42.00	--	--	2.04E+19	6.97E+19
Future[(b)]	--	48.00	--	--	2.31E+19	7.98E+19
Future[(b)]	--	54.00	--	--	2.59E+19	8.99E+19
Future[(b)]	--	60.00	--	--	2.86E+19	1.00E+20

Note(s):

(a) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

(b) Values beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-10 Calculated Iron Atom Displacement Rate and Iron Atom Displacements at the Surveillance Capsule Positions

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Displacement Rate (dpa/s)		Displacements (dpa)	
			4°	40°	4°	40°
1	1.08	1.08	3.38E-11	1.23E-10	1.15E-03	4.17E-03
2	0.91	1.99	3.70E-11	1.23E-10	2.22E-03	7.70E-03
3	1.23	3.22	3.54E-11	1.01E-10	3.59E-03	1.16E-02
4	0.92	4.14	3.54E-11	1.02E-10	4.62E-03	1.46E-02
5	1.11	5.25	3.63E-11	9.15E-11	5.89E-03	1.78E-02
6	1.17	6.42	3.14E-11	9.63E-11	7.05E-03	2.14E-02
7	1.12	7.54	3.46E-11	8.83E-11	8.27E-03	2.45E-02
8	1.12	8.66	2.73E-11	8.93E-11	9.23E-03	2.76E-02
9	1.16	9.82	2.56E-11	9.04E-11	1.02E-02	3.09E-02
10	1.14	10.96	2.11E-11	8.08E-11	1.09E-02	3.39E-02
11	1.23	12.20	2.06E-11	9.72E-11	1.17E-02	3.76E-02
12	1.40	13.60	1.98E-11	6.91E-11	1.26E-02	4.07E-02
13	1.04	14.64	1.96E-11	6.39E-11	1.33E-02	4.28E-02
14	1.18	15.82	2.39E-11	8.68E-11	1.41E-02	4.60E-02
15	1.32	17.14	2.18E-11	9.05E-11	1.51E-02	4.98E-02
16	1.35	18.49	2.33E-11	7.59E-11	1.60E-02	5.30E-02
17	1.35	19.84	2.13E-11	9.08E-11	1.69E-02	5.69E-02
18	1.37	21.21	2.43E-11	7.21E-11	1.80E-02	6.00E-02
19	1.28	22.48	2.42E-11	7.70E-11	1.90E-02	6.31E-02
20	1.40	23.89	2.22E-11	9.55E-11	2.00E-02	6.73E-02
21	1.31	25.20	2.23E-11	8.41E-11	2.09E-02	7.08E-02
22	1.40	26.60	2.16E-11	7.72E-11	2.18E-02	7.42E-02
23	1.14	27.74	2.01E-11	8.48E-11	2.26E-02	7.73E-02
24[(a)]	1.37	29.12	2.03E-11	9.05E-11	2.34E-02	8.12E-02
Future[(b)]	--	36.00	--	--	2.85E-02	1.01E-01
Future[(b)]	--	42.00	--	--	3.30E-02	1.18E-01
Future[(b)]	--	48.00	--	--	3.74E-02	1.35E-01
Future[(b)]	--	54.00	--	--	4.19E-02	1.52E-01
Future[(b)]	--	60.00	--	--	4.63E-02	1.69E-01

Note(s):

(a) Cycle 24 was the current operating cycle at the time these neutron exposures were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

(b) Values beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-11 Calculated Surveillance Capsule Lead Factors

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Lead Factor	
			4°	40°
1	1.08	1.08	0.96	3.33 (Capsule T)
2	0.91	1.99	1.01	3.34
3	1.23	3.22	1.08	3.33 (Capsule Y)
4	0.92	4.14	1.10	3.32
5	1.11	5.25	1.15	3.32 (Capsule X)
6	1.17	6.42	1.14	3.31
7	1.12	7.54	1.17	3.30
8	1.12	8.66	1.16	3.31 (Capsule U)
9	1.16	9.82	1.14	3.31
10	1.14	10.96	1.12	3.31
11	1.23	12.20	1.08	3.31
12	1.40	13.60	1.07	3.31
13	1.04	14.64	1.07	3.30
14	1.18	15.82	1.06	3.30
15	1.32	17.14	1.04	3.30
16	1.35	18.49	1.04	3.30
17	1.35	19.84	1.03	3.30
18	1.37	21.21	1.03	3.29
19	1.28	22.48	1.03	3.29
20	1.40	23.89	1.02	3.29
21	1.31	25.20	1.01	3.29
22	1.40	26.60	1.01	3.29
23	1.14	27.74	1.00	3.29
24[(a)]	1.37	29.12	0.99	3.29
Future[(b)]	--	36.00	0.97	3.29
Future[(b)]	--	42.00	0.96	3.28
Future[(b)]	--	48.00	0.95	3.28
Future[(b)]	--	54.00	0.94	3.27
Future[(b)]	--	60.00	0.94	3.27

Note(s):

(a) Cycle 24 was the current operating cycle at the time these lead factors were determined. Values listed for this cycle are projections based on the Cycle 24 design data.

(b) Values beyond Cycle 24 are based on the average core power distributions and reactor operating conditions of Cycles 21–23, but include a 1.1 bias on the core thermal power.

Table 2-12 Projected Fast Neutron Fluence Rate (E > 1.0 MeV) at the Surveillance Capsule Positions (Future Operation)

Capsule Position	Fluence Rate (n/cm^2-s)
4°	1.45E+10
40°	5.34E+10

Table 2-13 Calculational Uncertainties

Description	Uncertainty	
	Capsule	Vessel Inner Radius
PCA Comparisons	3%	3%
H.B. Robinson Comparisons	5%	5%
Analytical Sensitivity Studies	9%	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%	5%
Net Calculational Uncertainty	12%	13%

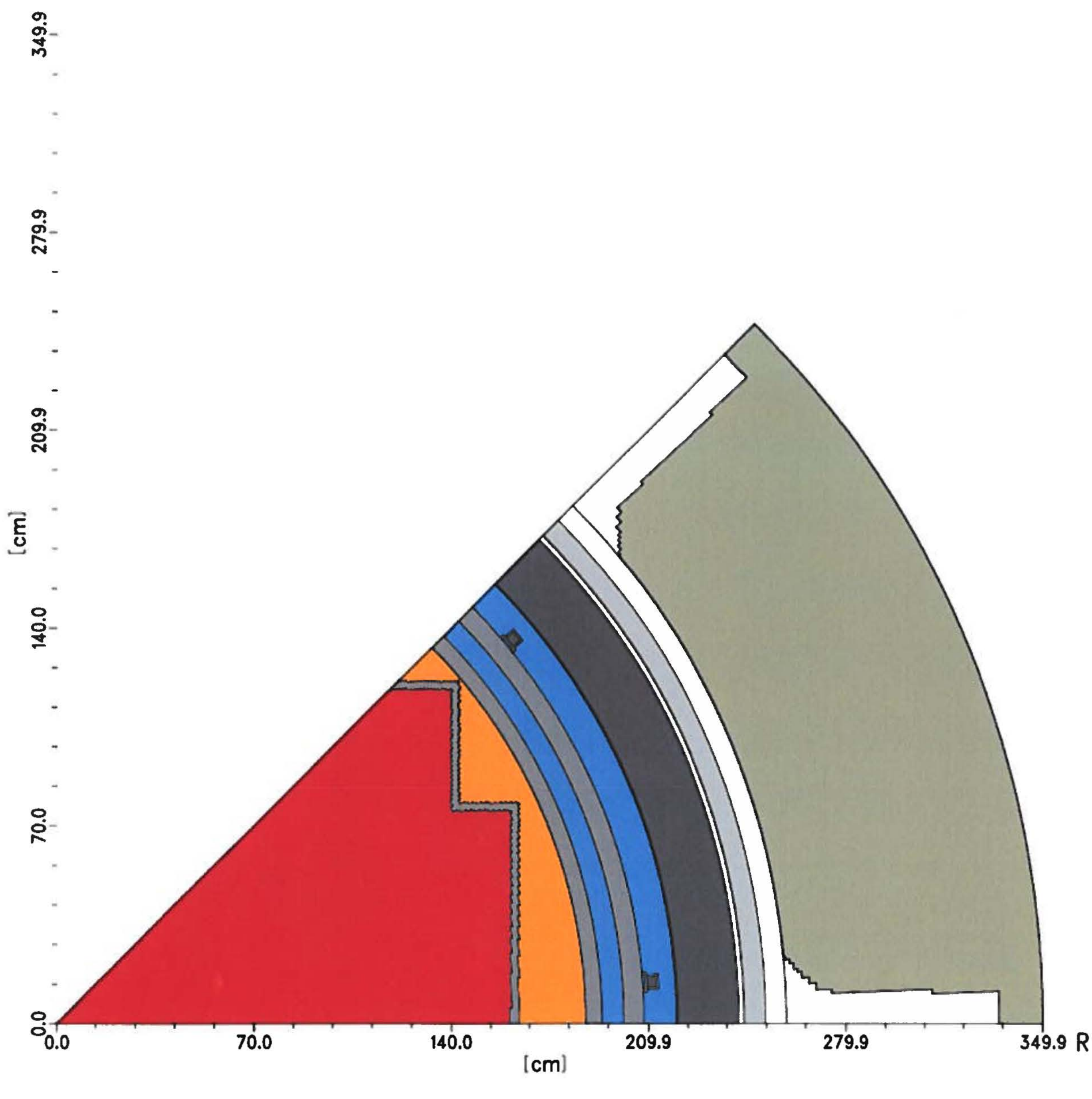

Figure 2-1 Plan View of the Reactor Geometry at the Core Midplane

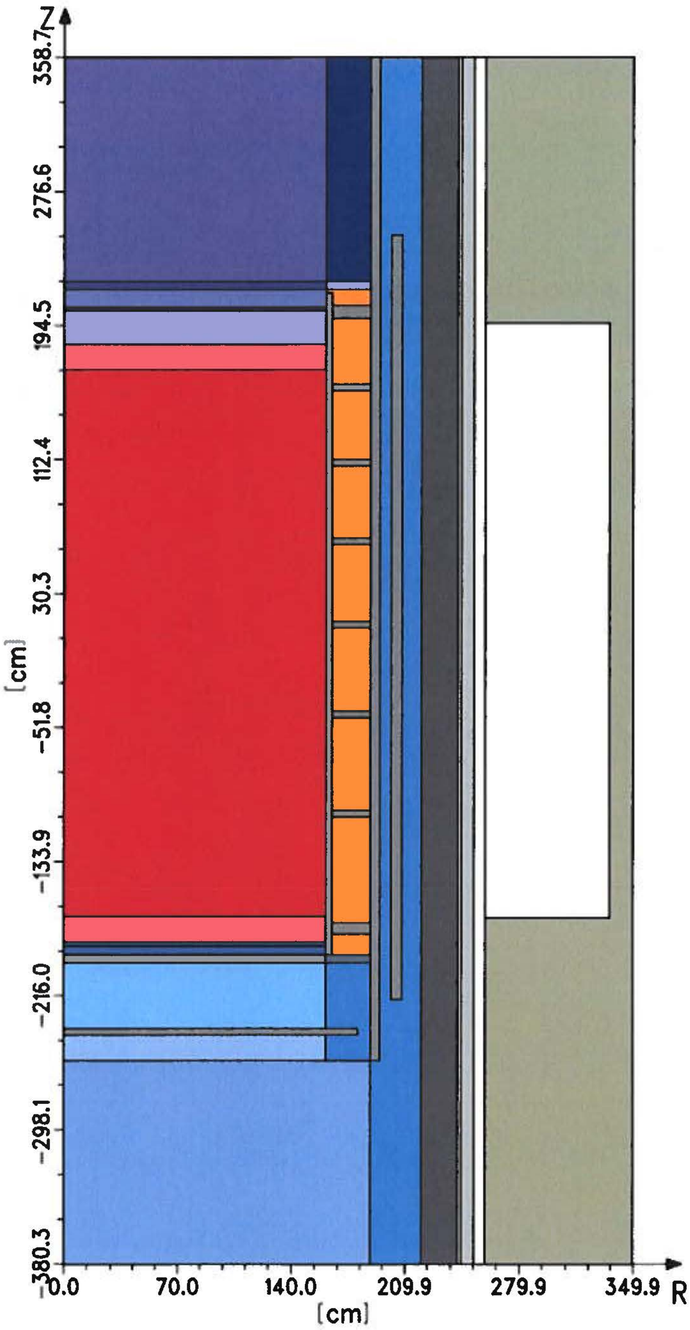

Figure 2-2 Section View of the Reactor Geometry – 0° Azimuth

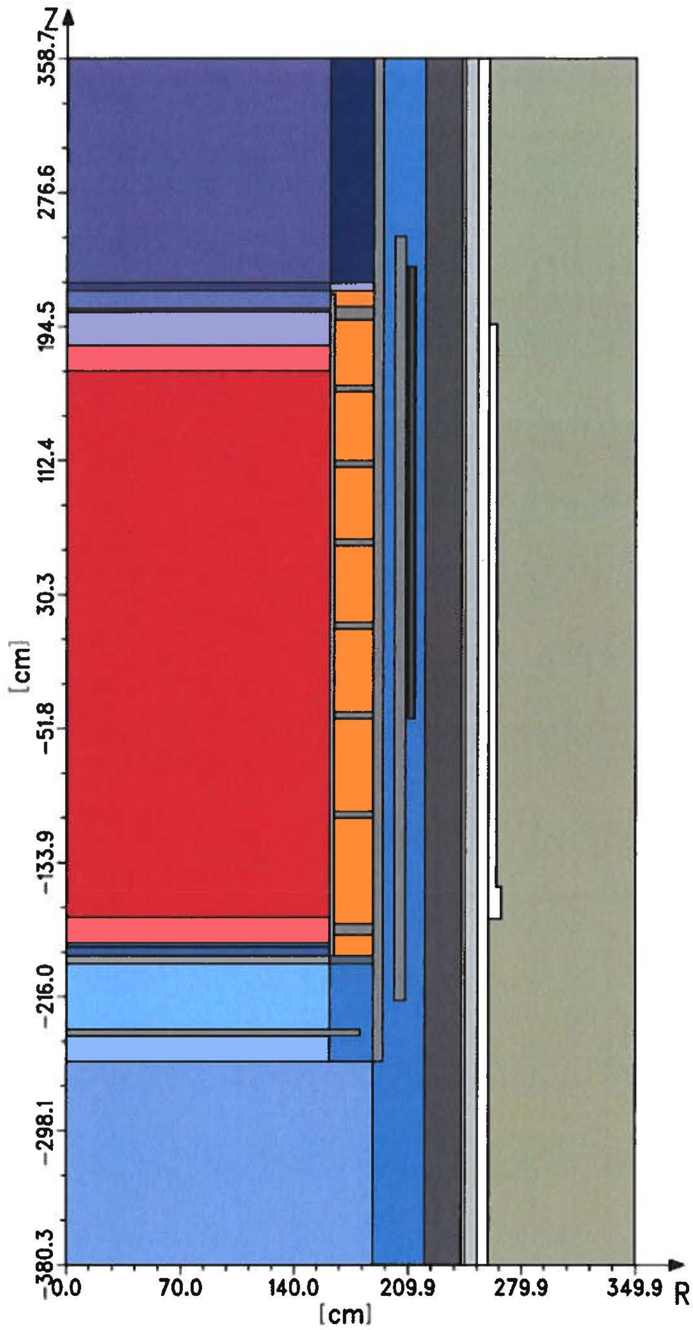

Figure 2-3 Section View of the Reactor Geometry – 4° Azimuth

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [4]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *"the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The D.C. Cook Unit 2 beltline materials traditionally included the Intermediate Shell Plates, Lower Shell Plates, the Intermediate to Lower Shell Circumferential Weld, the Intermediate Shell Longitudinal Welds, and the Longitudinal Shell Welds. However, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [10], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. The extended beltline materials include the Upper Shell Plates, Upper Shell Longitudinal Welds, and Upper Shell to Lower Shell Circumferential Weld. As seen from Table 2-7 of this report, the fluence for the both inlet/outlet nozzle to upper shell welds are less than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at 48 EFPY. Therefore, these materials do not need to be considered in the extended beltline.

A summary of the best-estimate copper (Cu) and nickel (Ni) contents, in units of weight percent (wt. %), as well as the initial RT_{NDT} values for the reactor vessel beltline, extended beltline, nozzle, and sister-plant materials are provided in Table 3-1 for D.C. Cook Unit 2. Table 3-2 provides the initial RT_{NDT} values for the replacement reactor vessel closure head and vessel flange materials for D.C. Cook Unit 2.

Table 3-1 Summary of the Best-Estimate Chemistry and Initial RT_{NDT} Values for the D.C. Cook Unit 2 Reactor Vessel Materials

Reactor Vessel Material	Heat Number	Wt. % Cu	Wt. % Ni	$RT_{NDT(u)}$ [c] (°F)
Reactor Vessel Beltline Materials[a]				
Intermediate Shell Plate 10-1	C5556-2	0.15	0.57	58
Intermediate Shell Plate 10-2	C5521-2	0.13	0.58	38
Lower Shell Plate 9-1	C5540-2	0.11	0.64	-20
Lower Shell Plate 9-2	C5592-1	0.14	0.59	-20
Intermediate Shell Longitudinal Welds	S3986 (Linde 124 flux, Lot # 934)	0.056	0.956	-35
Lower Shell Longitudinal Welds	S3986 (Linde 124 flux, Lot # 934)	0.056	0.956	-35
Intermediate Shell to Lower Shell Circumferential Weld	S3986 (Linde 124 flux, Lot # 934)	0.056	0.956	-35
D.C. Cook Unit 2 Surveillance Weld	S3986 (Linde 124 flux, Lot # 934)	0.055	0.97	---
Reactor Vessel Extended Beltline Materials[b]				
Upper Shell Plate 11-1	C5521-1	0.14	0.59	0
Upper Shell Plate 11-2	C5518-1	0.12	0.57	10
Upper Shell Plate 11-3	C5518-2	0.12	0.61	20
Upper Shell Longitudinal Welds	Note (d)	0.35	1.0	10[e]
Upper Shell to Intermediate Shell Circumferential Weld	S3986 (Linde 124 flux, Lot # 934)	0.056	0.956	-35
Reactor Vessel Nozzle Materials[g]				
Inlet Nozzle 269T-1	Q2Q8VW	Note (f)	Note (f)	-10[h]
Inlet Nozzle 269T-2	Q2Q8VW	Note (f)	0.85	-20
Inlet Nozzle 270T-1	Q2Q7VW	Note (f)	0.91	-20
Inlet Nozzle 270T-2	Q2Q7VW	Note (f)	Note (f)	0[h]
Outlet Nozzle 271T-1	Q2Q9VW	Note (f)	0.80	0
Outlet Nozzle 271T-2	Q2Q9VW	Note (f)	0.80	0
Outlet Nozzle 272T-1	Q2Q10VW	Note (f)	Note (f)	-10[h]
Outlet Nozzle 272T-2	Q2Q10VW	Note (f)	Note (f)	0[h]

Notes:

(a) All data taken from [12], unless noted.

(b) All data taken from [13], unless noted.

(c) The initial RT_{NDT} values are based on measured data for all beltline and extended beltline materials, unless otherwise noted. Thus, σ_I = 0°F.

(d) No data is available for the upper shell longitudinal welds. Per the guidance of 10 CFR 50.61 [17], 0.35 weight percent copper and 1.0 weight percent nickel are used.

(e) This is considered a generic value so a σ_I = 17°F is used.

(f) Weight-% Cu and Ni values were not reported in the certified material test report (CMTR). Generic wt. % Cu and Ni values for SA-508, Class 2 low-alloy steel are available in [14].

(g) All information for the inlet and outlet nozzle forgings is taken from CMTRs, and is based on measured data, unless otherwise noted.

(h) Consistent with [13], this is an approximation which uses measured data and a method similar to BWRVIP-173.

Table 3-2 Initial RT_{NDT} Values for the D.C. Cook Unit 2 Reactor Vessel Closure Head and Vessel Flange Materials

Reactor Vessel Material	Unit 2 Initial RT_{NDT} (°F)
Replacement Closure Head	-40[(a)]
Vessel Flange	30[(b)]

Notes:

(a) Data taken from [28].

(b) Data taken from [12].

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [1], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister plant' data.

The surveillance capsule plate material for D.C. Cook Unit 2 is from Intermediate Shell Plate 10-2. Since this material shares a Heat number (Heat # C5521) with Upper Shell Plate 11-1, the surveillance plate data is also applicable to this reactor vessel plate. Per Appendix C, the surveillance data are deemed credible for D.C. Cook Unit 2; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7 for the D.C. Cook Unit 2 Intermediate Shell Plate 10-2 and Upper Shell Plate 11-1.

The D.C. Cook Unit 2 surveillance weld specimens were fabricated from the reactor vessel Intermediate Shell to Lower Shell Circumferential Weld material (Heat # S3986). Since this material shares a Heat # (Heat # S3986) with all of the beltline and extended beltline longitudinal and circumferential welds, the surveillance weld data is also applicable to all of the welds. Per Appendix C, the surveillance data are deemed credible for D.C. Cook Unit 2; therefore, a reduced margin term will be utilized in the ART calculations contained in Section 7 for the D.C. Cook Unit 2 welds.

Table 4-1 summarizes the surveillance data available for the D.C. Cook Unit 2 plate and weld materials that will be used in the calculation of the Position 2.1 chemistry factor values.

Table 4-1 D.C. Cook Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift[b] (°F)
Intermediate Shell Plate 10-2 (Longitudinal)	T	0.246	55
	Y	0.685	90
	X	1.05	95
	U	1.63	95
Intermediate Shell Plate 10-2 (Transverse)	T	0.246	80
	Y	0.685	100
	X	1.05	103
	U	1.63	130
D. C. Cook Unit 2 Surveillance Weld (Heat # S3986)	T	0.246	40
	Y	0.685	50
	X	1.05	70
	U	1.63	75

Notes:

(a) Fluence values are from Table 2-9.

(b) Measured ΔRT_{NDT} values are from [12].

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using Regulatory Guide 1.99, Revision 2 [1], Positions 1.1 and 2.1. Position 1.1 CFs for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of [1]. The best-estimate copper and nickel weight percent values for the D.C. Cook Unit 2 reactor vessel materials are provided in Table 3-1 of this report.

The Position 2.1 CFs are calculated for the materials that have available surveillance program results. The calculation is performed using the method described in [1]. The D.C. Cook Unit 2 surveillance data are summarized in Section 4 of this report and will be utilized in the Position 2.1 CF calculations in this section. Table 5-1 and Table 5-2 calculate the D.C. Cook Unit 2 Position 2.1 CFs.

Position 1.1 and Position 2.1 CFs are summarized in Table 5-3 for D.C. Cook Unit 2. Adjustment of the ΔRT_{NDT} values were required per [1] due to chemistry differences between the surveillance plate (Intermediate Shell Plate 10-2) and Upper Shell Plate 11-1, and between the reactor vessel welds and the surveillance weld. The Position 1.1 CF for Intermediate Shell Plate 10-2 is 90.4°F, while the CF for Upper Shell Plate 11-1 is 99.6°F. The Position 1.1 CF for the reactor vessel welds is 76.4°F, while the surveillance program weld Position 1.1 CF is 75.0°F. Therefore, the chemistry adjustment factor is equal to 99.6 / 90.4 = 1.10 for Upper Shell Plate 11-1 and 76.4 / 75.0 = 1.02 for the welds. No temperature adjustments were undertaken since only D.C. Cook Unit 2 surveillance data is being considered.

Table 5-1 D.C. Cook Unit 2 Reactor Vessel Intermediate Shell Plate 10-2 and Weld Chemistry Factor Calculations Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm², E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate 10-2 (Longitudinal)	T	0.246	0.620	55	34.10	0.38
	Y	0.685	0.894	90	80.45	0.80
	X	1.05	1.014	95	96.30	1.03
	U	1.63	1.135	95	107.80	1.29
Intermediate Shell Plate 10-2 (Transverse)	T	0.246	0.620	80	49.60	0.38
	Y	0.685	0.894	100	89.39	0.80
	X	1.05	1.014	103	104.41	1.03
	U	1.63	1.135	130	147.52	1.29
				SUM:	709.57	7.00
	$CF_{10\text{-}2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (709.57) \div (7.00) =$ **101.4°F**					
Surveillance Weld Metal	T	0.246	0.620	40.8 (40)	25.29	0.38
	Y	0.685	0.894	51.0 (50)	45.59	0.80
	X	1.05	1.014	71.4 (70)	72.37	1.03
	U	1.63	1.135	76.5 (75)	86.81	1.29
				SUM:	230.07	3.50
	$CF_{Surveillance\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (230.07) \div (3.50) =$ **65.8°F**					

Notes:

(a) The capsule fluence values are from Table 4-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from [12]. Also note that the ΔRT_{NDT} values for the surveillance weld have been increased by a factor of 1.02 to account for the differences in weld chemistry. The actual measured values are shown in parenthesis.

Table 5-2 D.C. Cook Unit 2 Upper Shell Plate 11-1 Chemistry Factor Calculation Using Surveillance Capsule Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[b]	Measured ΔRT_{NDT}[a] (°F)	Adjusted ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Upper Shell Plate 11-1 (Longitudinal)	T	0.246	0.620	55	60.5	37.51	0.38
	Y	0.685	0.894	90	99.0	88.50	0.80
	X	1.05	1.014	95	104.5	105.93	1.03
	U	1.63	1.135	95	104.5	118.58	1.29
Upper Shell Plate 11-1 (Longitudinal)	T	0.246	0.620	80	88.0	54.56	0.38
	Y	0.685	0.894	100	110.0	98.33	0.80
	X	1.05	1.014	103	113.3	114.85	1.03
	U	1.63	1.135	130	143.0	162.27	1.29
					SUM:	780.52	7.00
	$CF_{11\text{-}1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (780.52) \div (7.00) =$ **111.5°F**						

Notes:

(a) The capsules fluence and ΔRT_{NDT} values are taken from Table 4-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) The adjusted ΔRT_{NDT} values have been increased by a factor of 1.10 to account for the differences in weld chemistry.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code [3]. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T-RT_{NDT})]} \quad (1)$$

where,

K_{Ic} (ksi√in.) = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

6.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

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

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

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \text{ x } (pR_i/t) \qquad (3)$$

Axial Flaw Methodology

For plates, forgings, and longitudinal welds, M_m for an inside axial surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

and, M_m for an outside axial surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

Circumferential Flaw Methodology

Similarly, for circumferential welds, M_m for an inside or an outside circumferential surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,
M_m = $0.443\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 1.53 for $\sqrt{t} > 3.464$

Where:

p = internal pressure (ksi), R_i = vessel inner radius (in), and t = vessel wall thickness (in.).

For bending stress, the corresponding K_I for the postulated axial or circumferential defect is:

$$K_{Ib} = M_b \text{ * Maximum Stress, where } M_b \text{ is two-thirds of } M_m \qquad (4)$$

The maximum K_I produced by radial thermal gradient for the postulated axial or circumferential inside surface defect of G-2120 is:

$$K_{It} = 0.953\text{x}10^{-3} \text{ x CR x } t^{2.5} \qquad (5)$$

where CR is the cooldown rate in °F/hr., or for a postulated axial or circumferential outside surface defect

$$K_{It} = 0.753\text{x}10^{-3} \text{ x HU x } t^{2.5} \qquad (6)$$

where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from

ASME Code, Section XI, Appendix G, Figure G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Code, Section XI, Appendix G, Figure G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a 1/4T axial or circumferential inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (7)$$

or similarly, K_{It} during heatup for a 1/4T outside axial or circumferential surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (8)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (9)$$

and x is a variable that represents the radial distance (in) from the appropriate (i.e., inside or outside) surface to any point on the crack front, and a is the maximum crack depth (in.).

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the P-T limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [2] Section 2.6 (Equations 2.6.2-4 and 2.6.3-1). Finally, the reactor vessel metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A, Revision 4 (Equation 2.6.1-1). This equation is solved utilizing values for thermal diffusivity of 0.518 ft^2/hr at 70°F and 0.379 ft^2/hr at 550°F and a constant convective heat-transfer coefficient value of 7000 Btu/hr-ft^2-°F.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw (the postulated flaw has a depth of 1/4 of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, Paragraph G-2120), the appropriate value for RT_{NDT}, and the reference fracture toughness curve (Equation 1). The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained, and from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4T flaw of Appendix G to Section XI of the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients, which increase with increasing cooldown rates, produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Since an inside surface flaw has a higher tensile stress than an outside flaw and is subject to more neutron embrittlement than an outside surface flaw in the beltline region, postulation of outside flaw for cooldown conditions is unnecessary. Allowable P-T curves are generated for steady-state (zero-rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) across the vessel wall developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location, and therefore, allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable P-T relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the inside 1/4T flaw during heatup is lower than the K_{Ic} for the flaw during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the P-T curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The third portion of the heatup analysis concerns the calculation of the P-T limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of P-T curves for the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the least of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

6.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [4] 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 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is calculated to be 621 psig. The initial RT_{NDT} values of the reactor vessel closure head and vessel flange are documented in Table 3-2. The limiting unirradiated RT_{NDT} of 30°F is associated with the vessel flange of the D.C. Cook Unit 2 reactor vessel, so the minimum allowable temperature of this region is 150°F at pressures greater than 621 psig (without margins for instrument uncertainties). This limit is shown in Figures 8-1 and 8-2.

6.4 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [4]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [2], the minimum boltup temperature should be 60°F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. Since the limiting unirradiated RT_{NDT} of this region is below 60°F per Table 3-2, the minimum boltup temperature for the D.C. Cook Unit 2 reactor vessel is 60°F. This limit is shown in Figure 8-1.

7 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (10)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [11]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \qquad (11)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth:

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \qquad (12)$$

where x inches (reactor vessel cylindrical shell beltline thickness is 8.5 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 11 to calculate the ΔRT_{NDT} at the specific depth.

The projected reactor vessel neutron fluence was updated for this analysis and documented in Section 2 of this report. The evaluation methods used in Section 2 are consistent with the methods presented in WCAP-14040-A, Revision 4 [2].

Table 7-1 contains the surface fluence values at 48 EFPY for D.C. Cook Unit 2. These values are used for the development of the P-T limit curves contained in this report. Table 7-1 also contains the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2 [1]. The values in this table are used to calculate the 48 EFPY ART values for the D.C. Cook Unit 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value. When a generic value is used, the σ_I is obtained from the set of data used to establish the mean. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings when surveillance data is not used or is non-credible, and 8.5°F (half the value) for plates or forgings when credible surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used or is non-credible, and is 14°F (half the value) when credible surveillance capsule data is used. Per [1], the value for σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 7-2 and 7-3 are the 48 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the D.C. Cook Unit 2 heatup and cooldown curves. Note that the longitudinal and circumferential beltline welds are combined into one calculation since each is fabricated from the same heat number and flux material. The limiting weld fluence, applicable to the circumferential beltline weld, is used for these ART calculations. The limiting ART values for D.C. Cook Unit 2 are summarized in Table 7-4.

Per Table 2-7, the inlet and outlet nozzle forgings and welds for D.C. Cook Unit 2 have projected fluence values at the lowest extent of the nozzle welds that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 48 EFPY. Consistent with NRC RIS 2014-11 [10], neutron radiation embrittlement need not be considered herein for either the inlet or outlet nozzle materials.

Table 7-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T, and 3/4T Locations for the D.C. Cook Unit 2 Reactor Vessel Beltline and Extended Beltline Materials at 48 EFPY

Reactor Vessel Region	Surface Fluence, f [(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	Surface FF	1/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f ($x10^{19}$ n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials						
Intermediate Shell Plate 10-1	2.44	1.240	1.47	1.106	0.528	0.822
Intermediate Shell Plate 10-2	2.44	1.240	1.47	1.106	0.528	0.822
Lower Shell Plate 9-1	2.42	1.238	1.45	1.104	0.524	0.819
Lower Shell Plate 9-2	2.42	1.238	1.45	1.104	0.524	0.819
Intermediate Shell Longitudinal Welds	1.00	1.000	0.600	0.857	0.217	0.589
Lower Shell Longitudinal Welds	0.753	0.920	0.452	0.779	0.163	0.522
Intermediate to Lower Shell Circumferential Weld	2.39	1.235	1.44	1.100	0.518	0.816
Reactor Vessel Extended Beltline Materials						
Upper Shell Plate 11-1	0.0126	0.128	0.00757	0.090	0.00273	0.042
Upper Shell Plate 11-2	0.0126	0.128	0.00757	0.090	0.00273	0.042
Upper Shell Plate 11-3	0.0126	0.128	0.00757	0.090	0.00273	0.042
Upper Shell Longitudinal Welds	0.0191[(b)]	0.167	0.0115	0.120	0.00414	0.058
Upper to Intermediate Shell Circumferential Weld	0.0191	0.167	0.0115	0.120	0.00414	0.058

Notes:

(a) 48 EFPY surface fluence values are documented in Table 2-7.

(b) The Upper Shell Longitudinal Welds fluence value is conservatively set equal to the Upper to Intermediate Shell Circumferential Weld fluence value.

Table 7-2 Adjusted Reference Temperature Evaluation for the D.C. Cook Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 48 EFPY at the 1/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF(a) (°F)	1/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	Margin (°F)	ART(e) (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Plate 10-1	1.1	108.4	1.47	1.106	58	119.9	0	17.0	34.0	211.9
Intermediate Shell Plate 10-2	1.1	90.4	1.47	1.106	38	100.0	0	17.0	34.0	172.0
Intermediate Shell Plate 10-2 Using Credible D.C. Cook Unit 2 Surveillance Data	2.1	101.4	1.47	1.106	38	112.1	0	8.5	17.0	167.1
Lower Shell Plate 9-1	1.1	74.6	1.45	1.104	-20	82.3	0	17.0	34.0	96.3
Lower Shell Plate 9-2	1.1	99.5	1.45	1.104	-20	109.8	0	17.0	34.0	123.8
Beltline Weld Seams (Heat # S3986)	1.1	76.4	1.44	1.100	-35	84.1	0	28.0	56.0	105.1
Beltline Weld Seams Using Credible D.C Cook Unit 2 Surveillance Data (Heat # S3986)	2.1	65.8	1.44	1.100	-35	72.4	0	14.0	28.0	65.4
Reactor Vessel Extended Beltline Materials										
Upper Shell Plate 11-1	1.1	99.6	0.00757	0.090	0	9.0	0	4.5	9.0	18.0
Upper Shell Plate 11-1 Using Credible D.C. Cook Unit 2 Surveillance Data	2.1	111.5	0.00757	0.090	0	10.1	0	5.0	10.1	20.2
Upper Shell Plate 11-2	1.1	82.4	0.00757	0.090	10	7.4	0	3.7	7.4	24.9
Upper Shell Plate 11-3	1.1	83.2	0.00757	0.090	20	7.5	0	3.8	7.5	35.0
Upper Shell Longitudinal Welds	1.1	272.0	0.0115	0.120	10	32.7	17	16.4	47.2	89.9
Upper to Intermediate Shell Circumferential Weld (Heat # S3986)	1.1	76.4	0.0115	0.120	-35	9.2	0	4.6	9.2	-16.6
Upper to Intermediate Shell Circumferential Weld Using Credible D.C. Cook Unit 2 Surveillance Data (Heat # S3986)	2.1	65.8	0.0115	0.120	-35	7.9	0	4.0	7.9	-19.2

Notes:

(a) Values are taken from Table 5-3.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix C, both the intermediate shell plate material surveillance data and the weld Heat # S3986 surveillance data are determined to be credible. Therefore, per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 data and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-3 Adjusted Reference Temperature Evaluation for the D.C. Cook Unit 2 Reactor Vessel Beltline and Extended Beltline Materials through 48 EFPY at the 3/4T Location

Reactor Vessel Material	R.G. 1.99, Rev. 2 Position	CF[a] (°F)	3/4T Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[b]	$RT_{NDT(U)}$[c] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[d] (°F)	Margin (°F)	ART[e] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Plate 10-1	1.1	108.4	0.528	0.822	58	89.1	0	17.0	34.0	181.1
Intermediate Shell Plate 10-2	1.1	90.4	0.528	0.822	38	74.3	0	17.0	34.0	146.3
Intermediate Shell Plate 10-2 Using Credible D.C. Cook Unit 2 Surveillance Data	2.1	101.4	0.528	0.822	38	83.3	0	8.5	17.0	138.3
Lower Shell Plate 9-1	1.1	74.6	0.524	0.819	-20	61.1	0	17.0	34.0	75.1
Lower Shell Plate 9-2	1.1	99.5	0.524	0.819	-20	81.5	0	17.0	34.0	95.5
Beltline Weld Seams (Heat # S3986)	1.1	76.4	0.518	0.816	-35	62.3	0	28.0	56.0	83.3
Beltline Weld Seams Using Credible D.C. Cook Unit 2 Surveillance Data (Heat # S3986)	2.1	65.8	0.518	0.816	-35	53.7	0	14.0	28.0	46.7
Reactor Vessel Extended Beltline Materials										
Upper Shell Plate 11-1	1.1	99.6	0.00273	0.042	0	4.2	0	2.1	4.2	8.4
Upper Shell Plate 11-1 Using Credible D.C. Cook Unit 2 Surveillance Data	2.1	111.5	0.00273	0.042	0	4.7	0	2.3	4.7	9.4
Upper Shell Plate 11-2	1.1	82.4	0.00273	0.042	10	3.5	0	1.7	3.5	16.9
Upper Shell Plate 11-3	1.1	83.2	0.00273	0.042	20	3.5	0	1.8	3.5	27.0
Upper Shell Longitudinal Welds	1.1	272.0	0.00414	0.058	10	15.8	17	7.9	37.5	63.3
Upper to Intermediate Shell Circumferential Weld (Heat # S3986)	1.1	76.4	0.00414	0.058	-35	4.4	0	2.2	4.4	-26.1
Upper to Intermediate Shell Circumferential Weld Using Credible D.C. Cook Unit 2 Surveillance Data (Heat # S3986)	2.1	65.8	0.00141	0.058	-35	3.8	0	1.9	3.8	-27.3

Notes:

(a) Values are taken from Table 5-3.

(b) Values are taken from Table 7-1.

(c) Values are taken from Table 3-1.

(d) Per Appendix C, both the intermediate shell plate material surveillance data and the weld Heat # S3986 surveillance data are determined to be credible. Therefore, per the guidance of Regulatory Guide 1.99, Revision 2 [1], the base metal σ_Δ = 17°F for Position 1.1 and σ_Δ = 8.5°F for Position 2.1 with credible surveillance data, and the weld metal σ_Δ = 28°F for the Position 1.1 data and σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} per regulatory guidance in [1].

(e) ART values are calculated in accordance with [1].

Table 7-4 Limiting ART Values for D.C. Cook Unit 2 at 48 EFPY[(a)]

Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)	Limiting Material
211.9	181.1	Intermediate Shell Plate 10-1

Note:

(a) Values are the limiting values from Tables 7-2 and 7-3.

8 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel cylindrical beltline region using the methods discussed in Sections 6 and 7 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [2].

The highest ART values for D.C. Cook Unit 2 correspond to the Intermediate Shell Plate 10-1. For P-T limit curve development, the limiting ART values are conservatively rounded up to the nearest whole number and increased by 3°F as shown in Table 8-1. This additional margin is added to account for potential future increases to D.C. Cook Unit 2 fluence projections.

Table 8-1 ART Values Used In P-T Limit Curve Development for D.C. Cook Unit 2 at 48 EFPY[(a)]

Limiting Material	Limiting 1/4T ART Value (°F)	Limiting 3/4T ART Value (°F)
Intermediate Shell Plate 10-1	215	185

Note:
(a) Values correspond to the limiting ART values in Table 7-4 rounded up to the nearest whole number and have an additional 3°F of margin added.

Figure 8-1 presents the limiting heatup curves without margins for possible instrumentation errors using a heatup rate of 60°F/hr applicable for 48 EFPY, with the flange requirements and using the "Axial Flaw" methodology. Figure 8-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, -20, -40, -60, and -100°F/hr applicable for 48 EFPY, with the flange requirements and using the "Axial Flaw" methodology. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G. All materials with an assumed circumferential flaw have lower ART values; therefore, the less conservative circumferential flaw methodology does not require consideration.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 8-1 and 8-2. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until P-T combinations are to the right of the criticality limit line shown in Figure 8-1 (heatup curve only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

$$1.5\ K_{Im} < K_{Ic} \qquad (13)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 6-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$ [see page 6-1 Equation (1)],

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies P-T limits for core operation in order to provide additional margin during actual power production. The P-T limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding P-T curve for heatup and cooldown calculated as described in Section 6 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the D.C. Cook Unit 2 reactor vessel at 48 EFPY is 275°F; this temperature value is calculated based on Equation (13). The vertical line drawn from these points on the P-T curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 8-1 and 8-2 define all of the above limits for ensuring prevention of non-ductile failure for the D.C. Cook Unit 2 reactor vessel for 48 EFPY without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 8-1 and 8-2 are presented in Tables 8-2, 8-3, and 8-4. Vacuum refill limits for the Reactor Coolant System (RCS) are included in Figures 8-1 and 8-2.

Nozzle P-T limit curves have previously been developed for D.C. Cook Unit 2 in [13] and compared to the D.C. Cook Unit 2 32 EFPY beltline P-T limit curves from [12]. The 32 EFPY beltline P-T limit curves were shown to be bounding compared to the nozzle P-T limit curves. These nozzle P-T limit curves from [13] remain applicable through 48 EFPY because the projected nozzle forging fluence is less than 1×10^{17} n/cm^2 at 48 EFPY and therefore embrittlement effects need not be considered consistent with RIS 2014-11 [10]. Since the 48 EFPY beltline P-T limit curves developed herein are based on higher ART values and produce more limiting pressure-temperature combinations than the 32 EFPY beltline curves from [12], the curves developed herein are also more limiting than the nozzle P-T limit curves in [13]. Therefore, the issue raised by RIS 2014-11 concerning the stresses associated with the geometry of the inlet and outlet nozzles has been addressed and it is concluded that the nozzle P-T limits remain non-bounding compared to the beltline P-T limits developed herein.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: D.C. Cook Unit 2 Intermediate Shell Plate 10-1 using Regulatory Guide 1.99 Position 1.1 data

LIMITING ART VALUES AT 48 EFPY: 1/4T, 215°F (Axial Flaw)
3/4T, 185°F (Axial Flaw)

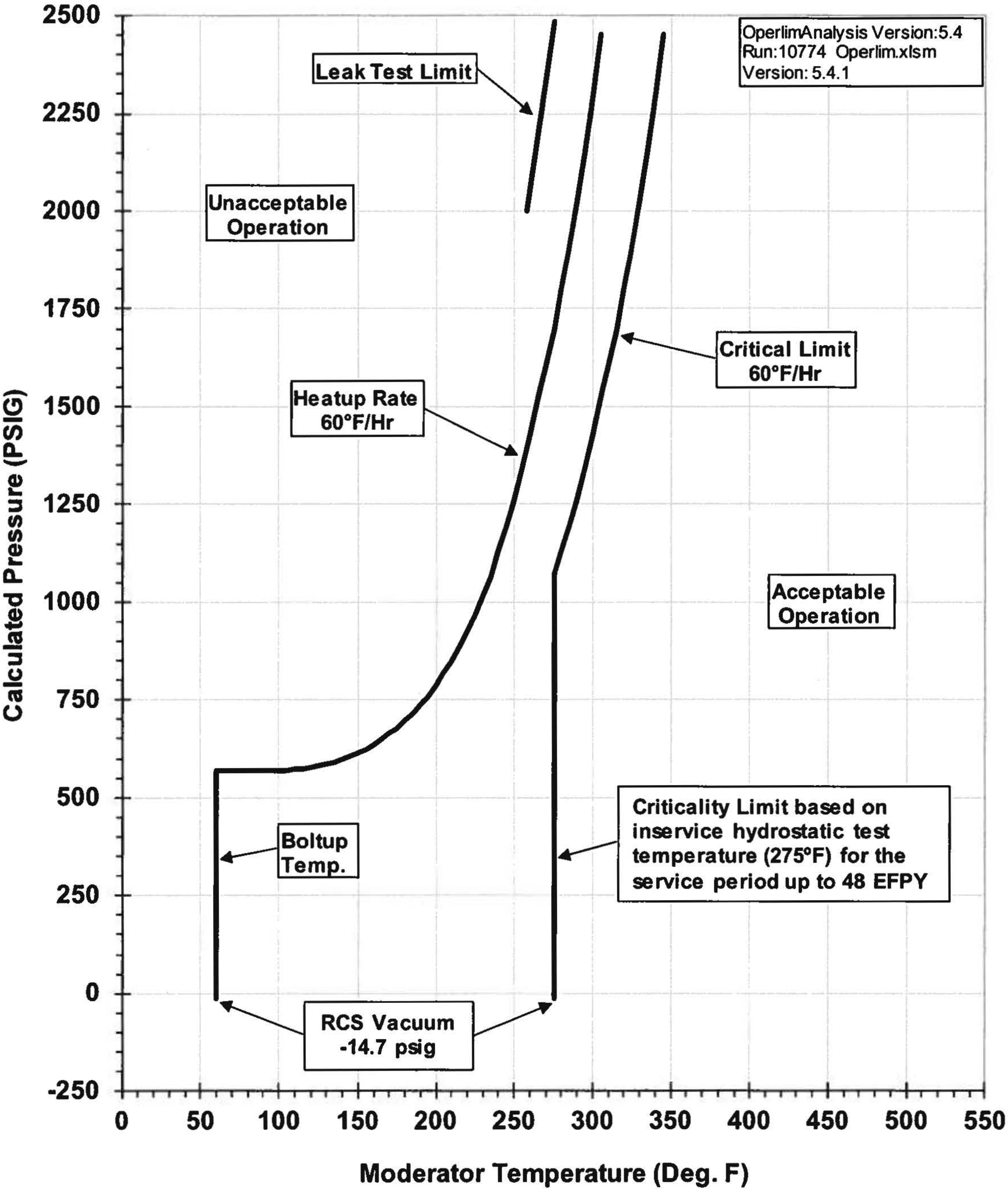

Figure 8-1 D.C. Cook Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for 48 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: D.C. Cook Unit 2 Intermediate Shell Plate 10-1 using Regulatory Guide 1.99 Position 1.1 data

LIMITING ART VALUES AT 48 EFPY: 1/4T, 215°F (Axial Flaw)
3/4T, 185°F (Axial Flaw)

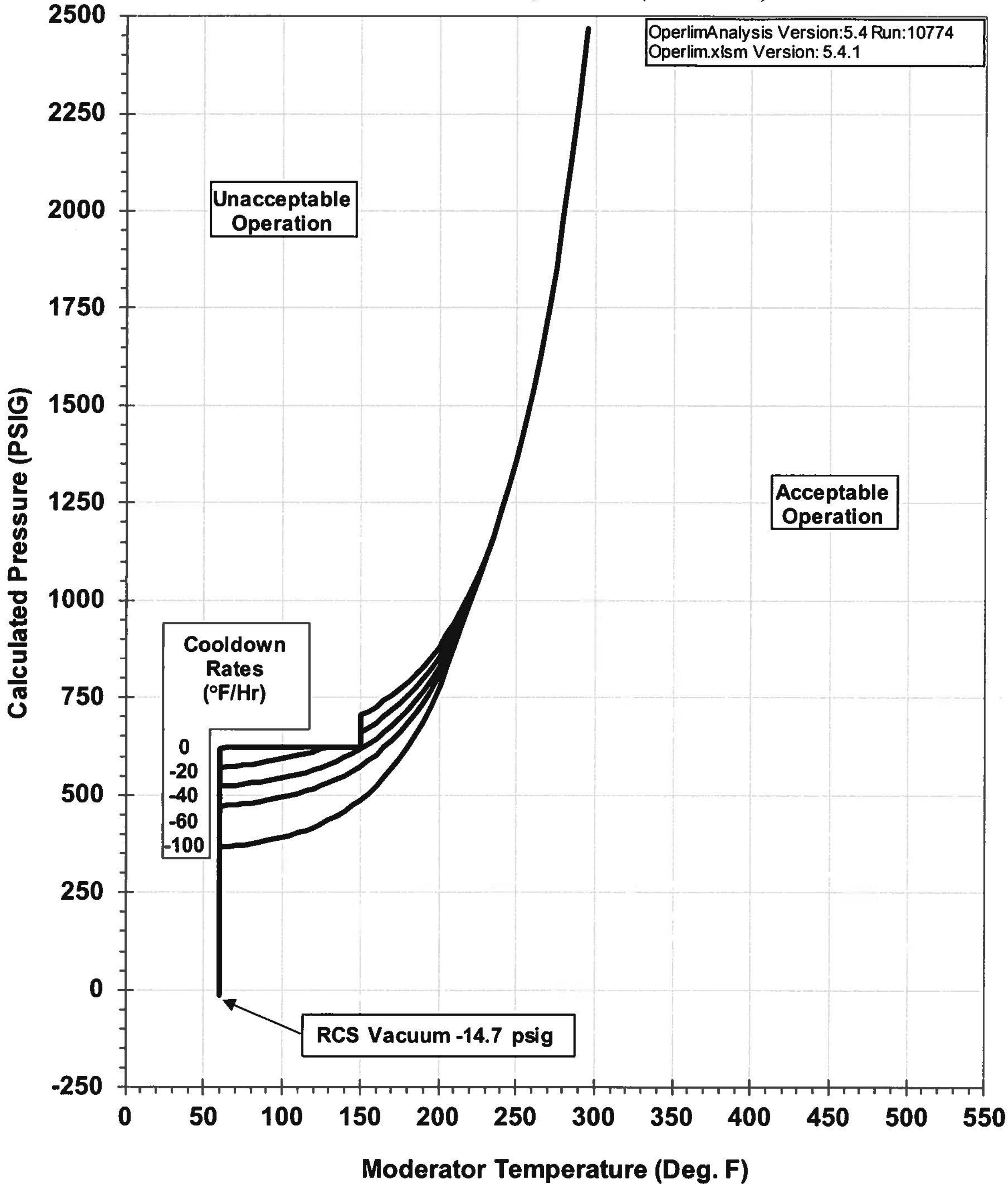

Figure 8-2 D.C. Cook Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, -20, -40, -60, and -100°F/hr) Applicable for 48 EFPY (<u>with</u> Flange Requirements and <u>without</u> Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-2 D.C. Cook Unit 2 48 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)
60	-14.7	275	-14.7
60	571	275	1067
65	571	280	1126
70	571	285	1191
75	571	290	1262
80	571	295	1340
85	571	300	1427
90	571	305	1523
95	571	310	1611
100	571	315	1698
105	571	320	1795
110	572	325	1902
115	574	330	2020
120	577	335	2150
125	581	340	2294
130	585	345	2452
135	591	-	-
140	598	-	-
145	606	-	-
150	615	-	-
155	625	-	-
160	636	-	-
165	649	-	-
170	663	-	-
175	679	-	-
180	696	-	-
185	715	-	-
190	737	-	-
195	761	-	-
200	787	-	-
205	816	-	-
210	848	-	-
215	884	-	-
220	923	-	-
225	967	-	-

60°F/hr Heatup		60°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)
230	1015	-	-
235	1067	-	-
240	1126	-	-
245	1191	-	-
250	1262	-	-
255	1340	-	-
260	1427	-	-
265	1523	-	-
270	1611	-	-
275	1698	-	-
280	1795	-	-
285	1902	-	-
290	2020	-	-
295	2150	-	-
300	2294	-	-
305	2452		

Table 8-3 D.C. Cook Unit 2 48 EFPY Leak Test Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Leak Test Limits	
T (°F)	P (psig)
258	2000
275	2485

Table 8-4 D.C. Cook Unit 2 48 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology for Steady-state (0°F/hr), -20°F/hr, -40°F/hr, -60°F/hr, and -100°F/hr (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		-20°F/hr Cooldown		-40°F/hr Cooldown		-60°F/hr Cooldown		-100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	-14.7	60	-14.7	60	-14.7	60	-14.7	60	-14.7
60	620	60	571	60	522	60	471	60	367
65	621	65	573	65	524	65	473	65	368
70	621	70	575	70	526	70	475	70	370
75	621	75	577	75	528	75	477	75	373
80	621	80	580	80	530	80	480	80	376
85	621	85	582	85	533	85	483	85	379
90	621	90	585	90	536	90	486	90	382
95	621	95	589	95	540	95	490	95	387
100	621	100	592	100	544	100	494	100	391
105	621	105	597	105	548	105	498	105	397
110	621	110	601	110	553	110	504	110	403
115	621	115	606	115	558	115	509	115	410
120	621	120	612	120	564	120	516	120	417
125	621	125	618	125	571	125	523	125	426
130	621	130	621	130	579	130	531	130	436
135	621	135	621	135	587	135	540	135	447
140	621	140	621	140	596	140	551	140	459
145	621	145	621	145	607	145	562	145	472
150	621	150	621	150	618	150	574	150	488
150	705	150	662	155	631	155	588	155	505
155	716	155	673	160	645	160	604	160	524
160	728	160	686	165	661	165	621	165	545
165	741	165	701	170	678	170	640	170	568
170	756	170	717	175	697	175	661	175	594
175	772	175	734	180	719	180	685	180	623
180	789	180	754	185	742	185	711	185	655

Steady-State		-20°F/hr Cooldown		-40°F/hr Cooldown		-60°F/hr Cooldown		-100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
185	809	185	775	190	769	190	740	190	691
190	831	190	799	195	798	195	772	195	731
195	855	195	825	200	830	200	808	200	775
200	881	200	855	205	866	205	848	205	824
205	911	205	887	210	905	210	891	210	879
210	943	210	923	215	949	215	940	215	939
215	979	215	962	220	997	220	994	220	994
220	1019	220	1006	225	1051	225	1051	225	1051
225	1062	225	1054	230	1108	230	1108	230	1108
230	1111	230	1108	235	1164	235	1164	235	1164
235	1164	235	1164	240	1223	240	1223	240	1223
240	1223	240	1223	245	1288	245	1288	245	1288
245	1288	245	1288	250	1361	250	1361	250	1361
250	1361	250	1361	255	1440	255	1440	255	1440
255	1440	255	1440	260	1528	260	1528	260	1528
260	1528	260	1528	265	1626	265	1626	265	1626
265	1626	265	1626	270	1733	270	1733	270	1733
270	1733	270	1733	275	1852	275	1852	275	1852
275	1852	275	1852	280	1984	280	1984	280	1984
280	1984	280	1984	285	2129	285	2129	285	2129
285	2129	285	2129	290	2289	290	2289	290	2289
290	2289	290	2289	295	2467	295	2467	295	2467
295	2467	295	2467	-	-	-	-	-	-

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APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) and vessel temperatures for the maximum heatup and cooldown rates at 48 EFPY for D.C. Cook Unit 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 88.845 inches
- 3/4T Radius = 93.095 inches

Table A-1 K_{It} and Vessel Temperature Values for D.C. Cook Unit 2 at 48 EFPY 60°F/hr Heatup Curves (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 60°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 60°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.443	-1.093	55.138	0.598
65	59.668	-2.491	55.806	1.642
70	63.128	-3.514	57.259	2.493
75	66.891	-4.424	59.374	3.200
80	70.899	-5.119	62.007	3.769
85	75.050	-5.719	65.080	4.238
90	79.389	-6.188	68.501	4.620
95	83.811	-6.594	72.207	4.935
100	88.364	-6.915	76.139	5.195
105	92.967	-7.195	80.254	5.413
110	97.658	-7.419	84.516	5.593
115	102.379	-7.617	88.897	5.747
120	107.159	-7.777	93.377	5.876
125	111.958	-7.921	97.934	5.987
130	116.797	-8.038	102.556	6.082
135	121.648	-8.147	107.230	6.165
140	126.525	-8.237	111.946	6.237
145	131.411	-8.322	116.696	6.301
150	136.313	-8.393	121.475	6.359
155	141.222	-8.462	126.276	6.411
160	146.140	-8.522	131.095	6.458
165	151.064	-8.580	135.929	6.502
170	155.994	-8.632	140.776	6.543
175	160.927	-8.684	145.633	6.582
180	165.864	-8.730	150.498	6.618
185	170.805	-8.777	155.369	6.654
190	175.747	-8.820	160.246	6.687
195	180.691	-8.864	165.127	6.720
200	185.636	-8.905	170.012	6.752
205	190.584	-8.947	174.899	6.783
210	195.531	-8.986	179.789	6.814

Table A-2 K_{It} and Vessel Temperature Values for D.C. Cook Unit 2 at 48 EFPY -100°F/hr Cooldown Curves (w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	236.226	16.489
205	231.142	16.422
200	226.058	16.356
195	220.974	16.289
190	215.889	16.222
185	210.805	16.155
180	205.720	16.088
175	200.635	16.021
170	195.550	15.954
165	190.465	15.887
160	185.380	15.820
155	180.294	15.753
150	175.209	15.686
145	170.124	15.618
140	165.039	15.552
135	159.954	15.485
130	154.869	15.418
125	149.783	15.351
120	144.699	15.285
115	139.614	15.218
110	134.529	15.152
105	129.444	15.085
100	124.360	15.019
95	119.275	14.953
90	114.191	14.888
85	109.107	14.822
80	104.023	14.756
75	98.939	14.690
70	93.855	14.625
65	88.771	14.560
60	83.689	14.494

APPENDIX B OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G [4] requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature (LST) requirement for all RCPB components, which is specified in NB-2332(b) and NB-3211 of the ASME Code, Section III [11], is the relevant requirement that would affect the P-T limits. This requirement is applicable to ferritic materials outside of the RV with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps and valves [11].

The D.C. Cook Unit 2 reactor coolant system does not have ferritic materials in the Class 1 piping, pumps, and valves (fabricated instead with stainless steel). Therefore, the LST requirements of the ASME Code, Section III, NB-2332(b) and NB-3211 [11] for these components do not need to be considered.

RIS 2014-11 [10] also addresses other ferritic components of the reactor coolant system relative to P-T limit, and states the following:

> *As specified in Sections I and IV.A of 10 CFR Part 50, Appendix G, ferritic RCPB components outside of the reactor vessel must meet the applicable requirements of ASME Code, Section III, "Rules for Construction of Nuclear Facility Components."*

The other ferritic RCPB components that are not part of the RV beltline or extended beltline for D.C. Cook Unit 2 consist of the RV closure head, steam generators, and pressurizer. The D.C. Cook Unit 2 primary system components are analyzed to the following ASME Code Section III Editions and met all applicable requirements at the time of construction. Therefore, no further consideration of these components is necessary.

- Replacement Reactor Vessel Closure Head – ASME Code Section III 1995 Edition through the 1996 Addenda
- Steam Generator – Portions of the original steam generators were replaced. The replacement steam generator components consist of the lower assemblies and the refurbished original components consist of the upper assemblies and internals (steam dome). The procurement of the replacement steam generator subassemblies did not affect the original design basis. Their designs are as follows:
  - Unit 2 Original Steam Generator Components - ASME Code Section III 1968 Edition through Winter 1968 Addenda
  - Unit 2 Replacement Steam Generator Components - ASME Code Section III 1983 Edition through Summer 1984 Addenda
- Pressurizer – ASME Code Section III 1965 Edition through Winter 1966 Addenda

APPENDIX C D.C. COOK UNIT 2 SURVEILLANCE PROGRAM CREDIBILITY EVALUATION

Regulatory Guide 1.99, Revision 2 [1] describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position 2.1 of [1], describes the method for calculating the adjusted reference temperature of reactor vessel beltline materials using surveillance capsule data. The methods of Position 2.1 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date, there have been four surveillance capsules removed and tested from the D.C. Cook Unit 2 reactor vessel. To use the surveillance data, the data must be shown to be credible. In accordance with [1], the credibility of the surveillance data will be judged based on five criteria.

The purpose of this evaluation is to apply the credibility requirements of [1], to the D.C. Cook Unit 2 reactor vessel surveillance data, including fluence values updated in Section 2, to determine if the surveillance data is credible.

C.1 D.C. COOK UNIT 2 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements" [4], as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

At the time of the design of the D.C. Cook Unit 2 surveillance program, the D.C. Cook Unit 2 reactor vessel beltline region was considered to consist of the following materials:

1. Intermediate Shell Plates 10-1 and 10-2 (Heat # C5556-2 and C5521-2)
2. Lower Shell Plates 9-1 and 9-2 (Heat # C5540-2 and C5592-1)
3. Intermediate Shell Axial Welds (Weld Wire Heat # S3986, Linde 124 Flux Type, Flux Lot # 934)
4. Lower Shell Axial Welds (Weld Wire Heat # S3986, Linde 124 Flux Type, Flux Lot # 934)
5. Intermediate Shell Plates to Lower Shell Plates Circumferential Weld Seam (Weld Wire Heat # S3986, Linde 124 Flux Type, Flux Lot # 934)

The D.C. Cook Unit 2 surveillance program utilizes longitudinal and transverse test specimens from the Intermediate Shell Plate 10-2 (Heat # C5521-2). These intermediate shell plates were chosen because they had higher initial RT_{NDT} values compared to the lower shell plates. The initial RT_{NDT} values, as well as the copper and phosphorus content, were determined to be equivalent for both intermediate shell plates. Intermediate Shell Plate 10-2 was chosen as the surveillance material due to the fact it has the lowest initial upper shelf energy (USE). It is noted that this plate is applicable to Upper Shell Plate 11-1, which shares the same heat number.

All D.C. Cook Unit 2 vessel beltline welds were fabricated with weld wire type ADCOM INMM, Heat # S3986, Linde 124 flux type, and Lot # 934. The surveillance weld was fabricated from the same weld wire heat and flux. Therefore, the surveillance weld metal is the same as all beltline welds and is representative of all beltline weld seams.

Therefore, the materials selected for use in the D.C. Cook Unit 2 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

<u>Based on the discussion above, Criterion 1 is met for the D.C. Cook Unit 2 surveillance program.</u>

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.

The credibility evaluation in [12] reviewed the plots of Charpy energy versus temperature for the unirradiated and irradiated conditions. This review concluded that, based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the D.C. Cook Unit 2 surveillance materials unambiguously.

Hence, the D.C. Cook Unit 2 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82 [15].

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for the weld and less than 17°F for the plate.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of [1]. The NRC methods were presented to industry at a meeting held by the NRC on February 12 and 13, 1998 [16]. At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance Data Available from Plant but No Other Source") will be used for the D.C. Cook Unit 2 surveillance data.

Following the NRC Case 1 guidelines, the D.C. Cook Unit 2 data will be evaluated. Table C-1 provides the calculation of the interim CFs for D.C. Cook Unit 2. Note that when evaluating the credibility of both the plate and weld data, the measured ΔRT_{NDT} values for the plate metal do not include the adjustment ratio procedure of Regulatory Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual plate metal measured shift values. In addition, only D.C Cook Unit 2 data is being considered; therefore, no temperature adjustment is required.

Table C-1 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using D.C. Cook Unit 2 Surveillance Data

Material	Capsule	Capsule Fluence[a] (x 10^{19} n/cm², E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Plate 10-2 (Longitudinal)	T	0.246	0.620	55	34.10	0.38
	Y	0.685	0.894	90	80.45	0.80
	X	1.05	1.014	95	96.30	1.03
	U	1.63	1.135	95	107.80	1.29
Intermediate Shell Plate 10-2 (Transverse)	T	0.246	0.620	80	49.60	0.38
	Y	0.685	0.894	100	89.39	0.80
	X	1.05	1.014	103	104.41	1.03
	U	1.63	1.135	130	147.52	1.29
				SUM:	709.57	7.00
	$CF_{10\text{-}2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (709.57) \div (7.00) =$ **101.4°F**					
Surveillance Weld Material (Heat # S3986)	T	0.246	0.620	40	24.80	0.38
	Y	0.685	0.894	50	44.70	0.80
	X	1.05	1.014	70	70.96	1.03
	U	1.63	1.135	75	85.11	1.29
				SUM:	225.56	3.50
	$CF_{Surveillance\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (225.56) \div (3.50) =$ **64.5°F**					

Notes:

(a) Taken from Table 4-1.
(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(c) Measured values are 30 ft-lb ΔRT_{NDT} values from [12].

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table C-2.

Table C-2 D.C. Cook Unit 2 Calculated Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF (Slope best-fit) (°F)	Capsule Fluence (x 10^{19} n/cm^2)	FF	Measured[(a)] ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	Scatter ΔRT_{NDT}[(b)] (°F)	<17°F (Base Metal) <28°F (Weld)
Intermediate Shell Plate 10-2 (Longitudinal)	T	101.4	0.246	0.620	55	62.9	7.9	Yes
	Y		0.685	0.894	90	90.6	0.6	Yes
	X		1.05	1.014	95	102.8	7.8	Yes
	U		1.63	1.135	95	115.1	20.1	No
Intermediate Shell Plate 10-2 (Transverse)	T		0.246	0.620	80	62.9	17.1	No
	Y		0.685	0.894	100	90.6	9.4	Yes
	X		1.05	1.014	103	102.8	0.2	Yes
	U		1.63	1.135	130	115.1	14.9	Yes
Surveillance Weld Metal (Heat # S3986)	T	64.5	0.246	0.620	40	40.0	0.0	Yes
	Y		0.685	0.894	50	57.7	7.7	Yes
	X		1.05	1.014	70	65.4	4.6	Yes
	U		1.63	1.135	75	73.2	1.8	Yes

Notes:

(a) Measured values are 30 ft-lb ΔRT_{NDT} values from [12].

(b) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

From a statistical point of view, ± 1σ would be expected to encompass 68% of the data. The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in [1], Position 2.1, should be less than 17°F for base metal and less than 28°F for weld metal. Table C-2 indicates that six of the eight surveillance data points fall inside the ± 1σ of 17°F scatter band for surveillance base metals (75% within the scatter band is greater than the 68% required to be credible); therefore, the plate data is deemed "credible" per the third criterion.

Table C-2 indicates that four of the four surveillance data points fall inside the ± 1σ of 28°F scatter band for surveillance weld metals (100% within the scatter band); therefore, the weld data is deemed "credible" per the third criterion.

Hence, Criterion 3 is met for the D.C. Cook Unit 2 surveillance program materials.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The D.C. Cook Unit 2 capsule specimens are located in the reactor between the thermal shield and the vessel wall and are positioned opposite the center of the core. The test capsules are in guide tubes attached to the thermal shields. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions and will not differ by more than 25°F.

Hence, Criterion 4 is met for the D.C. Cook Unit 2 surveillance program.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The D.C. Cook Unit 2 surveillance program does not correlation monitor material. Therefore, this criterion is not applicable to D.C. Cook Unit 2.

Conclusion: Based on the preceding responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B:

- The D.C. Cook Unit 2 surveillance plate data are deemed "credible"
- The D.C. Cook Unit 2 surveillance weld data are deemed "credible"

APPENDIX D VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

D.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for all surveillance capsules withdrawn from service to-date are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [5]. One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 2 of this report.

D.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the in-vessel neutron sensor sets withdrawn and analyzed to-date as part of the reactor vessel materials surveillance program are presented.

Eight irradiation capsules attached to the thermal shield were included in the reactor design to constitute the reactor vessel surveillance program. The capsules were located at azimuthal angles of 4° (Capsule S), 176° (Capsule V), 184° (Capsule W), and 356° (Capsule Z) that are 4° from the core cardinal axes and 40° (Capsule T), 140° (Capsule U), 220° (Capsule X), and 320° (Capsule Y) that are 40° from the core cardinal axes. The irradiation history of each of these eight in-vessel surveillance capsules is summarized as follows:

Capsule	Location	Irradiation History
T	40°	Cycle 1 (withdrawn for analysis)
Y	40°	Cycles 1–3 (withdrawn for analysis)
X	40°	Cycles 1–5 (withdrawn for analysis)
U	40°	Cycles 1–8 (withdrawn for analysis)
S	4°	In the reactor
V	4°	In the reactor
W	4°	In the reactor
Z	4°	In the reactor

The azimuthal locations included in the above tabulation represent the FOE azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of the surveillance capsules are summarized as follows:

Sensor Material	Reaction of Interest	Capsule T	Capsule Y	Capsule X	Capsule U
Copper	$^{63}Cu(n,\alpha)^{60}Co$	X	X	X	X
Iron	$^{54}Fe(n,p)^{54}Mn$	X	X	X	X
Nickel	$^{58}Ni(n,p)^{58}Co$	X	X	X	X
Uranium-238	$^{238}U(n,f)^{137}Cs$	X	X	X	X
Neptunium-237	$^{237}Np(n,f)^{137}Cs$	X	X	X	X
Cobalt-Aluminum*	$^{59}Co(n,\gamma)^{60}Co$	X	X	X	X
*The cobalt-aluminum measurements include both bare wire and cadmium-covered sensors.					

Pertinent physical and nuclear characteristics of the in-vessel surveillance capsule passive neutron sensors are listed in Table D-1.

The use of passive monitors such as those listed above does not yield a direct measure of the energy-dependent neutron fluence rate at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron fluence rate has on the target material over the course of the irradiation period. An accurate assessment of the average neutron fluence rate incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- The measured specific activity of each monitor,
- The physical characteristics of each monitor,
- The operating history of the reactor,
- The energy response of each monitor, and
- The neutron energy spectrum at the monitor location.

Results from the radiometric counting of the neutron sensors from the in-vessel capsules are documented in [18–21], and re-evaluated in this appendix using the RAPTOR-M3G model described in Section 2. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples. In the case of the uranium and neptunium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by the in-vessel capsules was based on monthly power generation data from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The startup and shutdown dates for each cycle of operation used in the evaluations are given in Table D-2.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 \cdot F \cdot Y \cdot \sum \frac{P_j}{P_{ref}} \cdot C_j \cdot \left(1 - e^{-\lambda \cdot t_j}\right) \cdot e^{-\lambda \cdot t_{d,j}}}$$

where:

R	=	Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
A	=	Measured specific activity (dps/g).
N_0	=	Number of target element atoms per gram of sensor.
F	=	Atom fraction of the target isotope in the target element.
Y	=	Number of product atoms produced per reaction.
P_j	=	Average core power level during irradiation period j (MW).
P_{ref}	=	Maximum or reference power level of the reactor (MW).
C_j	=	Calculated ratio of ϕ (E > 1.0 MeV) during irradiation period j to the time weighted average ϕ (E > 1.0 MeV) over the entire irradiation period.
λ	=	Decay constant of the product isotope (1/sec).
t_j	=	Length of irradiation period j (sec).
$t_{d,j}$	=	Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variations of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology described in Section 2, accounts for the change in sensor reaction rates caused by variations in fluence rate induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low-leakage fuel management, the additional C_j term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel-cycle-specific neutron fluence rate values are used to compute cycle-dependent C_j values at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, additional corrections were made to the ^{238}U measurements to account for the presence of ^{235}U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the ^{238}U and ^{237}Np sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the fission sensor reaction rates are summarized as follows:

Correction	Capsule T	Capsule Y	Capsule X	Capsule U
^{235}U Impurity/Pu Build-in	0.875	0.858	0.844	0.821
$^{238}U(\gamma,f)$	0.958	0.958	0.958	0.958
Net ^{238}U Correction	0.837	0.822	0.808	0.787
$^{237}Np(\gamma,f)$	0.984	0.984	0.985	0.985

These factors were applied in a multiplicative fashion to the decay-corrected uranium and neptunium fission sensor reaction rates.

Results of the sensor reaction rate determinations for the in-vessel capsules are given in Table D-3 through Table D-6.

D.1.2 Least-Squares Evaluation of Sensor Sets

Least-squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations, resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best estimates for key exposure parameters such as ϕ (E > 1.0 MeV) or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties. For example:

$$R_i \pm \delta_{R_i} = \sum_g \left(\sigma_{ig} \pm \delta_{\sigma_{ig}}\right) \cdot \left(\sigma_g \pm \delta_{\sigma_g}\right)$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the surveillance capsule dosimetry, the FERRET Code [22] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters (ϕ (E > 1.0 MeV) and dpa) along with associated uncertainties for the in-vessel capsules analyzed to-date.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the plant-specific application of the least-squares methodology, the calculated neutron spectrum was obtained from the results of the neutron transport calculations described in Section 2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section D.1.1. The dosimetry reaction cross sections and uncertainties were obtained from the Sandia National Laboratories Radiation Metrology Laboratory (SNLRML) dosimetry cross-section library [23]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018-09, "Application of ASTM Evaluated Cross-Section Data File, Matrix E706 (IIB)" [24].

The uncertainties associated with the measured reaction rates, dosimetry cross sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIIA)" [25].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the surveillance capsule sensor sets withdrawn and analyzed to-date.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)^{137}Cs$	10%
$^{237}Np(n,f)^{137}Cs$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections were compiled from the most recent cross-section evaluations, and they have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the plant-specific reactor vessel surveillance program, the following uncertainties in the fission spectrum averaged cross sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08–4.16%
$^{54}Fe(n,p)^{54}Mn$	3.05–3.11%
$^{58}Ni(n,p)^{58}Co$	4.49–4.56%
$^{238}U(n,f)^{137}Cs$	0.54–0.64%
$^{237}Np(n,f)^{137}Cs$	10.32–10.97%
$^{59}Co(n,\gamma)^{60}Co$	0.79–3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectra input to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

Using the uncertainties associated with the reaction rates obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random group-wise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1 - \theta]\delta_{gg'} + \theta\, e^{-H}$$

where

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when $g = g'$ and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

D.1.3 Comparisons of Measurements and Calculations

This section provides comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations. These comparisons are provided on two levels. In the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. In the second level, calculated values of neutron exposure rates in terms of fast neutron fluence rate ϕ (E > 1.0 MeV) and iron atom displacement rate are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table D-7, comparisons of M/C ratios are listed for the threshold sensors contained in the in-vessel capsules. From Table D-7, it is noted that for the individual threshold sensors, the average M/C ratio ranges from 0.91 to 1.04 with an overall average of 0.96 and an associated standard deviation of 7.9%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table D-8, best-estimate-to-calculation (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate resulting from the least-squares evaluation of each dosimetry set. For the in-vessel capsules, the average BE/C ratio is 0.93 with an associated uncertainty of 4.9% for fast neutron fluence rate (E > 1.0 MeV) and 0.94 with an associated uncertainty of 4.1% for the iron atom displacement rate.

The M/C comparisons based on individual sensor reactions without recourse to the least-squares adjustment procedure are summarized as follows:

Reaction	In-Vessel Capsules	
	Avg. M/C	% Unc. (1σ)
^{63}Cu(n,α)	1.04	4.3%
^{54}Fe(n,p)	0.93	7.8%
^{58}Ni(n,p)	0.97	5.7%
^{238}U(Cd)(n,f)	0.91	7.9%
^{237}Np(Cd)(n,f)	0.98	9.0%
Linear Average	0.96	7.9%

A similar comparison for exposure rate expressed in terms of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s) are summarized as follows:

Parameter	In-Vessel Capsules	
	Avg. M/C	% Unc. (1σ)
Fast Neutron Fluence Rate (E > 1.0 MeV)	0.93	4.9%
Iron Atom Displacement Rate (dpa/s)	0.94	4.1%

These data comparisons show similar and consistent results, with the linear average M/C ratio of 0.96 in good agreement with the resultant least-squares BE/C ratios of 0.93 for neutron fluence rate (E > 1.0 MeV) and 0.94 for iron atom displacement rate. The comparisons demonstrate that the calculated results provided in Section 2 of this report are validated within the context of the assigned 13% uncertainty and, further, show that the ±20% (1σ) agreement between calculation and measurement required by [5] is met.

Table D-1 Nuclear Parameters Used in the Evaluation of the In-Vessel Surveillance Capsule Neutron Sensors

Reaction of Interest	Atomic Weight[(a)] (g/g-atom)	Target Atom Fraction[(b),(c)]	Product Half-life[(b),(c),(d)] (days)	Fission Yield[(d)] (%)
^{63}Cu (n,α) ^{60}Co	63.546	0.6917	1925.28	n/a
^{54}Fe (n,p) ^{54}Mn	55.845	0.05845	312.13	n/a
^{58}Ni (n,p) ^{58}Co	58.6934	0.68077	70.86	n/a
^{238}U (n,f) ^{137}Cs	238.051	1.00	10975.76	6.02
^{237}Np (n,f) ^{137}Cs	237.048	1.00	10975.76	6.27
^{59}Co (n,γ) ^{60}Co	58.933	0.0015	1925.28	n/a

Note(s):

(a) Atomic weight data were taken from the Chart of the Nuclides, 17^{th} Edition, dated 2010 [26].

(b) Half-life and target atom fraction data for ^{63}Cu (n,α), ^{54}Fe (n,p), and ^{58}Ni (n,p), reactions were taken from ASTM Standard E1005-16 [27].

(c) The half-life for the ^{59}Co (n,γ) reaction was taken from ASTM Standard E1005-16 [27]. The target atom fractions for the ^{59}Co (n,γ), ^{238}U (n,f), and ^{237}Np (n,f) reactions are reflective of standard Westinghouse surveillance capsule dosimeter values.

(d) Half-life and fission yield data for the ^{238}U (n,f) and ^{237}Np (n,f) reactions were taken from ASTM Standard E1005-16 [27].

Table D-2 Startup and Shutdown Dates

Cycle	Startup Date	Shutdown Date
1	03/17/1978	10/19/1979
2	01/19/1980	03/14/1981
3	05/20/1981	11/21/1982
4	01/22/1983	03/10/1984
5	07/10/1984	02/28/1986
6	07/10/1986	04/23/1988
7	03/16/1989	06/30/1990
8	11/10/1990	02/22/1992

Table D-3 Measured Sensor Activities and Reaction Rates for Surveillance Capsule T

Sample ID	Target Isotope	Description	Measured Activity[(a)] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
1	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	Top Mid	4.53E+04	3.32E+05	5.06E-17	5.02E-17	5.02E-17
2	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	Mid	4.43E+04	3.25E+05	4.95E-17		
3	$^{63}Cu\ (n,\alpha)\ ^{60}Co$	Bot Mid	4.51E+04	3.31E+05	5.04E-17		
4	$^{54}Fe\ (n,p)\ ^{54}Mn$	Top	1.59E+06	3.16E+06	5.01E-15	5.05E-15	5.05E-15
5	$^{54}Fe\ (n,p)\ ^{54}Mn$	Top Mid	1.62E+06	3.22E+06	5.11E-15		
6	$^{54}Fe\ (n,p)\ ^{54}Mn$	Mid	1.57E+06	3.12E+06	4.95E-15		
7	$^{54}Fe\ (n,p)\ ^{54}Mn$	Bot Mid	1.63E+06	3.24E+06	5.14E-15		
8	$^{54}Fe\ (n,p)\ ^{54}Mn$	Bot	1.60E+06	3.18E+06	5.04E-15		
9	$^{58}Ni\ (n,p)\ ^{58}Co$	Top Mid	3.59E+07	5.02E+07	7.19E-15	7.12E-15	7.12E-15
10	$^{58}Ni\ (n,p)\ ^{58}Co$	Mid	3.50E+07	4.90E+07	7.01E-15		
11	$^{58}Ni\ (n,p)\ ^{58}Co$	Bot Mid	3.57E+07	5.00E+07	7.15E-15		
12	$^{238}U(Cd)\ (n,f)\ ^{137}Cs$	Mid	1.06E+05	4.33E+06	2.85E-14	2.85E-14	2.38E-14
13	$^{237}Np(Cd)\ (n,f)\ ^{137}Cs$	Mid	8.22E+05	3.36E+07	2.11E-13	2.11E-13	2.08E-13
14	$^{59}Co\ (n,\gamma)\ ^{60}Co$	Top	4.96E+06	3.67E+07	2.40E-12	2.72E-12	2.72E-12
15	$^{59}Co\ (n,\gamma)\ ^{60}Co$	Bot	6.31E+06	4.67E+07	3.05E-12		
16	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	Top (Cd)	Not Recovered	--	--	--	--
17	$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	Bot (Cd)	Not Recovered	--	--		

Note(s):

(a) Measured activities are decay corrected to October 19, 1979.

Table D-4 Measured Sensor Activities and Reaction Rates for Surveillance Capsule Y

Sample ID	Target Isotope	Description	Measured Activity[(a)] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
1	^{63}Cu (n,α) ^{60}Co	Top Mid	9.36E+04	2.80E+05	4.27E-17	4.32E-17	4.32E-17
2	^{63}Cu (n,α) ^{60}Co	Mid	9.59E+04	2.87E+05	4.38E-17		
3	^{63}Cu (n,α) ^{60}Co	Bot Mid	9.46E+04	2.83E+05	4.32E-17		
4	^{54}Fe (n,p) ^{54}Mn	Top	1.76E+06	2.68E+06	4.26E-15	4.37E-15	4.37E-15
5	^{54}Fe (n,p) ^{54}Mn	Top Mid	1.80E+06	2.74E+06	4.35E-15		
6	^{54}Fe (n,p) ^{54}Mn	Mid	1.84E+06	2.81E+06	4.45E-15		
7	^{54}Fe (n,p) ^{54}Mn	Bot Mid	1.83E+06	2.79E+06	4.43E-15		
8	^{54}Fe (n,p) ^{54}Mn	Bot	1.81E+06	2.76E+06	4.38E-15		
9	^{58}Ni (n,p) ^{58}Co	Top Mid	2.87E+07	4.43E+07	6.34E-15	6.38E-15	6.38E-15
10	^{58}Ni (n,p) ^{58}Co	Mid	2.88E+07	4.44E+07	6.36E-15		
11	^{58}Ni (n,p) ^{58}Co	Bot Mid	2.92E+07	4.50E+07	6.45E-15		
12	^{238}U(Cd) (n,f) ^{137}Cs	Mid	3.00E+05	4.26E+06	2.80E-14	2.80E-14	2.30E-14
13	^{237}Np(Cd) (n,f) ^{137}Cs	Mid	1.78E+06	2.53E+07	1.59E-13	1.59E-13	1.56E-13
14	^{59}Co (n,γ) ^{60}Co	Top	1.02E+07	3.08E+07	2.01E-12	2.01E-12	2.01E-12
15	^{59}Co (n,γ) ^{60}Co	Bot	1.02E+07	3.08E+07	2.01E-12		
16	^{59}Co(Cd) (n,γ) ^{60}Co	Top (Cd)	Not Recovered	--	--	--	--
17	^{59}Co(Cd) (n,γ) ^{60}Co	Bot (Cd)	Not Recovered	--	--		

Note(s):

(a) Measured activities are decay corrected to November 21, 1982.

Table D-5 Measured Sensor Activities and Reaction Rates for Surveillance Capsule X

Sample ID	Target Isotope	Description	Measured Activity[a] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
1	^{63}Cu (n,α) ^{60}Co	Top Mid	1.20E+05	2.79E+05	4.25E-17	4.28E-17	4.28E-17
2	^{63}Cu (n,α) ^{60}Co	Mid	1.20E+05	2.79E+05	4.25E-17		
3	^{63}Cu (n,α) ^{60}Co	Bot Mid	1.22E+05	2.83E+05	4.32E-17		
4	^{54}Fe (n,p) ^{54}Mn	Top	1.38E+06	2.68E+06	4.26E-15	4.31E-15	4.31E-15
5	^{54}Fe (n,p) ^{54}Mn	Top Mid	1.41E+06	2.74E+06	4.35E-15		
6	^{54}Fe (n,p) ^{54}Mn	Mid	1.40E+06	2.72E+06	4.32E-15		
7	^{54}Fe (n,p) ^{54}Mn	Bot Mid	1.42E+06	2.76E+06	4.38E-15		
8	^{54}Fe (n,p) ^{54}Mn	Bot	1.37E+06	2.67E+06	4.23E-15		
9	^{58}Ni (n,p) ^{58}Co	Top Mid	1.84E+07	4.26E+07	6.09E-15	6.06E-15	6.06E-15
10	^{58}Ni (n,p) ^{58}Co	Mid	1.81E+07	4.19E+07	5.99E-15		
11	^{58}Ni (n,p) ^{58}Co	Bot Mid	1.84E+07	4.26E+07	6.09E-15		
12	^{238}U(Cd) (n,f) ^{137}Cs	Mid	3.76E+05	3.42E+06	2.24E-14	2.24E-14	1.81E-14
13	^{237}Np(Cd) (n,f) ^{137}Cs	Mid	3.14E+06	2.85E+07	1.79E-13	1.79E-13	1.76E-13
14	^{59}Co (n,γ) ^{60}Co	Top	1.55E+07	3.60E+07	2.35E-12	2.33E-12	2.33E-12
15	^{59}Co (n,γ) ^{60}Co	Bot	1.52E+07	3.53E+07	2.30E-12		
16	^{59}Co(Cd) (n,γ) ^{60}Co	Top (Cd)	6.48E+06	1.80E+07	1.18E-12	1.18E-12	1.18E-12
17	^{59}Co(Cd) (n,γ) ^{60}Co	Bot (Cd)	Not Recovered	--	--		

Note(s):

(a) Measured activities are decay corrected to February 28, 1986.

Table D-6 Measured Sensor Activities and Reaction Rates for Surveillance Capsule U

Sample ID	Target Isotope	Description	Measured Activity[a] (Bq/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
1	^{63}Cu (n,α) ^{60}Co	Top Mid	1.23E+05	2.56E+05	3.90E-17	3.91E-17	3.91E-17
2	^{63}Cu (n,α) ^{60}Co	Mid	1.23E+05	2.56E+05	3.90E-17		
3	^{63}Cu (n,α) ^{60}Co	Bot Mid	1.24E+05	2.58E+05	3.94E-17		
4	^{54}Fe (n,p) ^{54}Mn	Top	9.19E+05	2.23E+06	3.54E-15	3.55E-15	3.55E-15
5	^{54}Fe (n,p) ^{54}Mn	Top Mid	9.45E+05	2.30E+06	3.64E-15		
6	^{54}Fe (n,p) ^{54}Mn	Mid	9.50E+05	2.31E+06	3.66E-15		
7	^{54}Fe (n,p) ^{54}Mn	Bot Mid	8.59E+05	2.09E+06	3.31E-15		
8	^{54}Fe (n,p) ^{54}Mn	Bot	9.30E+05	2.26E+06	3.59E-15		
9	^{58}Ni (n,p) ^{58}Co	Top Mid	3.93E+06	3.64E+07	5.22E-15	5.23E-15	5.23E-15
10	^{58}Ni (n,p) ^{58}Co	Mid	3.93E+06	3.64E+07	5.22E-15		
11	^{58}Ni (n,p) ^{58}Co	Bot Mid	3.96E+06	3.67E+07	5.26E-15		
12	^{238}U(Cd) (n,f) ^{137}Cs	Mid	5.91E+05	3.55E+06	2.33E-14	2.33E-14	1.83E-14
13	^{237}Np(Cd) (n,f) ^{137}Cs	Mid	4.26E+06	2.56E+07	1.60E-13	1.60E-13	1.58E-13
14	^{59}Co (n,γ) ^{60}Co	Top	1.88E+07	3.91E+07	2.55E-12	2.46E-12	2.46E-12
15	^{59}Co (n,γ) ^{60}Co	Bot	1.74E+07	3.62E+07	2.36E-12		
16	^{59}Co(Cd) (n,γ) ^{60}Co	Top (Cd)	Not Recovered	--	--	1.19E-12	1.19E-12
17	^{59}Co(Cd) (n,γ) ^{60}Co	Bot (Cd)	7.32E+06	1.82E+07	1.19E-12		

Note(s):

(a) Measured activities are decay corrected to August 24, 1992.

Table D-7 Comparison of Measured and Calculated Threshold Foil Reaction Rates for the In-Vessel Capsules

Reaction	Capsule				Average	Std. Dev.
	T	Y	X	U		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.10	1.01	1.04	1.00	1.04	4.3%
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.00	0.92	0.96	0.83	0.93	7.8%
$^{58}Ni\ (n,p)\ ^{58}Co$	1.02	0.98	0.98	0.89	0.97	5.7%
$^{238}U\ (n,f)\ ^{137}Cs$	0.95	0.98	0.82	0.88	0.91	7.9%
$^{237}Np\ (n,f)\ ^{137}Cs$	1.06	0.86	1.03	0.98	0.98	9.0%
Average of M/C Results					0.96	7.9%

Table D-8 Comparison of Calculated and Best-Estimate Exposure Rates for the In-Vessel Capsules

Capsule	Fast (E > 1.0 MeV) Fluence Rate		Iron Atom Displacement Rate	
	BE/C	Std. Dev.	BE/C	Std. Dev.
T	0.99	6.0%	0.99	7.0%
Y	0.92	6.0%	0.92	7.0%
X	0.93	6.0%	0.94	7.0%
U	0.88	6.0%	0.90	7.0%
Average	0.93	4.9%	0.94	4.1%

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Enclosure 7 to AEP-NRC-2021-28

LTR-SCS-20-18-NP, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," dated June 30, 2020 (Non-Proprietary)

Westinghouse

To: John T. Ahearn

Date: June 30, 2020

From: Functional, Systems & Setpoints Engineering

Phone: 412-374-4063

Our ref: LTR-SCS-20-18-NP
Revision 0

Subject: D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY

References:

1. CN-SCS-20-3, Revision 0, "D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," June 2020.

2. WCAP-18456-NP, Revision 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," February 2020.

This letter transmits the non-proprietary version of the Low Temperature Overpressure Protection System (LTOPS) analysis report for D.C. Cook Unit 2 to AEP. The detailed analysis was performed in Reference 1 using Pressure-Temperature (P-T) limits from Reference 2 that are applicable through the 60-year end of license extension corresponding to 48 effective full-power years (EFPY). The Westinghouse Non-Proprietary Class 3 version is provided in Attachment 1 with the proprietary information identified and redacted within brackets. A proprietary version of this letter will be issued separately with the information within brackets identified.

The attachment to this letter will be transmitted to AEP along with an application for withholding proprietary information from public disclosure and supporting affidavit. The types of proprietary information are identified via superscripts following each bracket, which correspond to the types described in item 5 of the corresponding application for withholding proprietary information.

For any questions, please contact the undersigned.

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Attachment 1: D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY (Non-Proprietary)

**Electronically approved records are authenticated in the electronic document management system.*

LTR-SCS-20-18-NP
Revision 0

D.C. Cook Unit 2 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY

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Functional and Systems Engineering

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Functional and Systems Engineering

1.0 Introduction

The Low Temperature Overpressure Protection System (LTOPS) provides Reactor Coolant System (RCS) pressure relief capability during low temperature operation (i.e., Modes 4, 5, and 6) to minimize the potential for challenging reactor vessel integrity limits (i.e., 10 CFR 50, Appendix G limits) when operating at low temperature conditions. At D.C. Cook, in accordance with Technical Specification LCO 3.4.12, the pressurizer Power Operated Relief Valves (PORVs), with reduced lift settings, and/or the Residual Heat Removal (RHR) suction relief valve provide a method of LTOP for the potential overpressure transients. The LTOPS PORV setpoints are selected in accordance with the NRC approved methodology in Reference 5 such that the peak pressure during the design basis Mass Injection (MI) and Heat Injection (HI) transients will not exceed the isothermal Appendix G Pressure-Temperature (P-T) limits.

Updated P-T limits have been developed for D.C. Cook Unit 2 through the 60-year end of license extension (EOLE) period, which are valid for operation through 48 EFPY (Reference 1). Therefore, an LTOPS analysis was performed in Reference 2 to establish the LTOPS configuration, relief valve setpoints, and operating limitations necessary to protect the revised P-T limits at 48 EFPY. This scope of work was proposed in Reference 10.

1.1 Limits of Applicability

The results of this report are applicable to D.C. Cook Unit 2 with the 60-year EOLE P-T limits defined in Reference 1 valid up to 48 EFPY and the key analysis inputs defined in Section 2.0.

1.2 Open Items

None.

2.0 Input Parameters and Assumptions

Key input parameters for the analysis were requested in Reference 3 and the AEP response was provided in Reference 4. The key input parameters and analysis assumptions are summarized as follows.

2.1 Key Inputs

<u>Design Basis MI Transient</u>

Currently, the D.C. Cook technical specifications define separate LTOP relief requirements depending whether one or two charging pumps are capable of injecting. Per Reference 4, AEP requested for this to be simplified for this analysis by defining a single set of LTOPS requirements that can accommodate the MI resulting from two charging pumps injecting. The design basis MI flow rate as a function of cold leg (or RCS) pressure confirmed in Reference 4 is shown in Table 1.

Table 1: MI Flow Rate vs. RCS Pressure

a,c

The MI transient was analyzed to develop PORV setpoint overshoots and undershoots for MI flow rates of [ ]a,c gpm with PORV setpoints ranging from [ ]a,c psig. This range of flow rates is adequate to cover the design basis MI transient at D.C. Cook Unit 2. The results of the MI transient parametric analyses are summarized in Section 5.1.

<u>Design Basis HI Transient</u>

The HI transient is defined as the startup of one RCP with the SG secondary side a maximum of 50°F hotter than each of the RCS cold leg temperatures. Prior to the RCP start, all loops are inactive and the entire RCS primary side (except for stagnant water in the SG tubes) is assumed to be 50°F cooler than the secondary side. For this analysis, RCS/SG temperatures of 60/110°F, 100/150°F, 150/200°F, 200/250°F, 250/300°F, and 300/350°F were analyzed to bound the range of temperatures applicable to LTOP. The results of the HI transient parametric analyses are summarized in Section 5.1.

Wide Range Pressure and Temperature Uncertainties

In accordance with the Reference 5 methodology, pressure and temperature uncertainties were applied during the development of the LTOPS PORV setpoints. The wide range temperature and pressure uncertainties were provided in Reference 4 as follows.

- Pressure uncertainty = [ ]^{a,c} psi
- Temperature uncertainty = [ ]^{a,c} °F

Pressure Drop between Reactor Vessel and Pressure Transmitter

The following values were calculated in Reference 6 and were confirmed valid for this analysis in Reference 4.

- For 4 RCPs running = [ ]^{a,c} psid
- For 2 RCPs running = [ ]^{a,c} psid
- For 1 RCP running = [ ]^{a,c} psid

D.C. Cook currently utilizes the following restrictions on RCP operation:

- Zero RCPs may be operating at RCS Temperatures (T_{RCS}) < 100°F
- No more than one RCP may be operating at 100°F $\leq T_{RCS} \leq$ 140°F
- All four RCPs are allowed to operate at T_{RCS} > 140°F

The LTOPS analysis performed herein conservatively assumes all four RCPs operating for the MI transient. The conditions for the design basis HI transient can only be established with zero RCPs running; then, the transient is initiated by starting one RCP. Therefore, by definition, the maximum ΔP that needs to be accounted for during the design basis HI transient is that associated with one RCP operating. The analysis does not impose or credit any restrictions on the number of RCPs that are allowed to be running.

Pressurizer PORV Characteristics, Stroke, and Delay Times

The pressurizer PORV characteristics and the stroke and delay times were provided in Reference 4.

- Valve full open C_V = [ ]^{a,c} gpm/$\sqrt{psi}$
- Opening Stroke time = [ ]^{a,c} seconds
- Closing Stroke Time = [ ]^{a,c} seconds
- Delay Time = [ ]^{a,c} seconds
- PORV backpressure = [ ]^{a,c} psig

RHR Suction Relief Valve Characteristics

The following RHR System characteristics were confirmed in Reference 4 unless otherwise noted.

- Nominal lift setting = 450 psig (Reference 4 and LCO 3.4.12)
- Setting tolerance = 3%
- Full open pressure = 495 psig (i.e., 10 % accumulation)
- Valve capacity = Table 2 (see Reference 9)
- RHR system design pressure = 600 psig
- RHR system analytical pressure limit = 660 psig (110 % of design pressure)
- RHR pump head [ ]a,c (Note 1)
- Pressurizer Relief Tank (PRT) backpressure = 100 psig prior to disk rupture, 0 psig after disk rupture
- Pressure drop from the reactor vessel to RHR relief valve inlet = [ ]a,c psi (Note 2)
- Autoclosure interlock status = Deleted

Notes:

1. Reference 4 determined that the RHR pump head is [ ]b,c psi, which is less than the maximum acceptable value of [ ]a,c psi.
2. Reference 9 determined that the pressure drop in the line from the RHR piping to the suction relief valve is [ ]a,c ft at a flow rate of [ ]a,c gpm. This equates to [ ]a,c psi for 60°F water. The assumed flow rate of [ ]a,c gpm conservatively bounds the maximum RHR suction relief valve capacity of [ ]a,c gpm for an inlet pressure of [ ]a,c psig and zero backpressure (Table 2). This pressure drop in the relief valve inlet piping needs to be added to a pressure drop from the RHR piping back to the reactor vessel. Since the wide range pressure transmitters are located at the RHR connection to the hot leg, the pressure drops calculated in Reference 6 are representative of this additional pressure drop. Therefore, assuming all four RCPs in operation, the maximum pressure drop from the reactor vessel midplane to the RHR suction relief valve inlet is [ ]a,c psi ([ ]a,c psi ΔP in relief valve inlet line + [ ]a,c psi from RV midplane to RHR connection).

Table 2: RHR Suction Relief Valve Capacity

a,c

Appendix G Limits

The steady-state (isothermal) Appendix G limits for D.C. Cook Unit 2 applicable for 48 EFPY are summarized in Table 3. Per Reference 5, steady-state P-T limits are used for LTOPS setpoint analysis. Note that the limits shown in Table 3 do not include instrumentation uncertainties; however, these uncertainties are included in the setpoint development as shown in Section 5.2.

Table 3: Steady-State Appendix G Limits D.C. Cook Unit 2 for 48 EFPY

D.C. Cook Unit 2 for 48 EFPY			
RCS Temperature (°F)	**Appendix G Limit (psig)**	**RCS Temperature (°F)**	**Appendix G Limit (psig)**
60	620	160	728
65	621	165	741
70	621	170	756
75	621	175	772
80	621	180	789
85	621	185	809
90	621	190	831
95	621	195	855
100	621	200	881
105	621	205	911
110	621	210	943
115	621	215	979
120	621	220	1019
125	621	225	1062
130	621	230	1111
135	621	235	1164
140	621	240	1223
145	621	245	1288
150	621	250	1361
150.1	705	300	2663
155	716	-	-

Pressurizer PORV Piping limit

In addition to the Appendix G limits, an 800 psig pressure limit is included to address pressurizer PORV piping loading considerations associated with subcooled water discharge. This limit is recognized as an operational consideration that is accommodated by the LTOPS in Reference 5. The PORV piping has been generically evaluated for the water hammer loads associated with cyclic water relief at up to 800 psig. Therefore, when the plant is operated water solid, the LTOPS settings ensure that the pressure does not exceed the design value of 800 psig.

2.2 Key Assumptions

The following key assumptions are applicable for the D.C. Cook Unit 2 LTOPS analysis:

1. It is assumed that the RCS is enclosed by a non-yielding, inelastic boundary. The pressurizer is assumed to be in a water solid condition with the water at the same subcooled temperature as the remainder of the RCS. [
   ]^{a,c}.
2. Only one PORV was credited to mitigate the low temperature overpressure event to meet the single failure criteria.
3. All MI cases are analyzed at an RCS temperature of 60°F, which is the minimum RCS temperature corresponding to the bolt up temperature (Reference 1). [
   ]^{a,c}.
4. For the HI transient, the entire RCS primary side, with the exception of the water in the SG tubes, is conservatively assumed to initially be 50°F cooler than the SG secondary side temperatures in all four SGs.
5. A single-phase, sub-cooled water discharge through the PORV was assumed.
6. Letdown flow is conservatively assumed to be isolated during the MI and HI transients. [
   ]^{a,c}.
7. The PORV C_V as a function of lift is assumed to vary linearly.
8. It is assumed that a pressurizer steam bubble will exist for operation at temperatures above the LTOPS arming temperature of 291°F.

3.0 Description of Analyses and Evaluations

The LTOPS pressure relief capabilities are provided by two pressurizer PORVs, with reduced lift settings, and/or the RHR suction relief valve. Alternately, a sufficiently sized RCS vent can provide protection when the RCS is depressurized. When the pressurizer PORVs are credited for LTOP, two PORVs are required to be operable, but only a single PORV is credited in the analysis to accommodate a single active failure (Reference 12, item 3). The RHR relief valve is a spring loaded water relief valve that does not require any actuation signals or motive power to operate. They are thus defined as passive components and are not subject to single active failures.

The RHR suction relief valve and/or pressurizer PORVs are required to mitigate the potential overpressure events that can occur during relatively low temperature RCS operation to ensure that both the reactor vessel Appendix G limits and the RHR piping limit are protected. The potential overpressure transients that must be considered consist of MI and HI transients defined in Section 2.1.

The LTOPS PORV setpoints are determined using the NRC approved methodology in Reference 5. Parametric analyses of the design basis MI and HI transients are performed using the LOFTRAN code (Reference 15). The purpose of these parametric analyses is to generate the transient pressure response data consisting of the PORV setpoint overshoot and undershoots. The LTOPS PORV setpoints are calculated based upon the PORV setpoints overshoot and undershoot data and the LTOPS setpoint acceptance criteria described in Section 4.0.

References 7 and 8 describe the basis and methodology for using the RHR suction relief valve to provide LTOP. Per Reference 4, the autoclosure interlock has been removed from the RHR suction isolation valve at D.C. Cook. Therefore, the RHR system cannot be spuriously isolated from the RCS, thus making the RHR relief valve a suitable option to provide LTOP. The Reference 2 analyses have been performed to demonstrate the acceptability of the RHR setting and capacity to provide protection against the design basis LTOP transients.

The RHR suction relief valve is guaranteed to achieve full capacity at 110% of the nominal set pressure (i.e., 495 psig). As long as the RHR suction relief valve capacity meets or exceeds the required relief capacity during the design basis LTOPS transients, the pressure at the inlet to the relief valve will not exceed 495 psig. Therefore, if the RHR suction relief valve has sufficient capacity, the peak pressure at the reactor vessel mid-plane will be the 495 psig accumulation pressure plus the applicable pressure drop back to the vessel. Similarly, the peak pressure in the RHR system will be the 495 psig accumulation pressure plus the RHR pump head.

For the mass injection transient, evaluating the RHR relief capacity consists of a straightforward comparison of the pump curve to the RHR relief valve capacity. For the heat injection transient, the required relief capacity corresponds to the fluid expansion rate. The fluid expansion rate during the HI transient is dependent upon of the RCS conditions at the initiation of the heat injection transient. Therefore, to determine the required relief capacity during the HI transient, the transient will be analyzed for a range of initial RCS conditions to determine the required relief capacity for the transient.

4.0 Acceptance Criteria

The following acceptance criteria from Reference 5 are used to determine the LTOPS PORV setpoints:

1. The peak RCS pressure resulting from the design basis MI and HI transients shall not exceed the minimum of the steady-state adjusted Appendix G limits and the PORV piping limit.
2. The minimum RCS pressure resulting from the design basis MI and HI transients should not drop below the RCP No. 1 Seal ΔP limit.

If there is a conflict between satisfying the upper limits (i.e., the minimum of the Appendix G limits and the piping limit) and the lower limits (i.e., the RCP No. 1 Seal ΔP limit), the upper pressure limits will take precedence. Furthermore, since D.C. Cook Unit 2 sets both PORVs to the same LTOPS setting, both PORVs will open during a best estimate LTOP event. Since the analyses herein only credit a single PORV, the resultant minimum RCS pressures following an LTOP actuation will be lower during a best estimate LTOP event than those shown in Tables 4 and 5. Therefore, criterion #2 may be challenged following LTOP actuations. An RCP No. 1 seal ΔP limit violation is not a nuclear safety issue. The RCP seals are designed to withstand momentary or incidental contact. Redundant plant indicators provide information as to the health of the seal, and any damage would be detected, and the plant operator could take the necessary corrective actions.

The following acceptance criteria are used to demonstrate the acceptability of the RHR suction relief valve:

1. The peak pressure in the reactor vessel must not exceed the Appendix G limit.

   This criterion is conservatively met if the installed valve capacity exceeds the required capacity for the design basis MI and HI transients with consideration of the applicable pressure drop from the reactor vessel to the RHR suction relief valve inlet.

2. The pressure at the RHR pump discharge must not exceed 110% of the RHR system design pressure (i.e., 660 psig, Reference 4).

   This criterion is conservatively met if the installed valve capacity exceeds the required capacity for the limiting MI and HI transients, with consideration of the RHR pump head.

5.0 Results and Conclusions

5.1 PORV Setpoint Overshoots and Undershoots

The pressure overshoot and undershoot are defined as the peak pressure minus the assumed PORV setpoint and the assumed PORV setpoint minus the minimum pressure during the transient, respectively.

Table 4 shows the summary of overshoots and undershoots for the MI flow rate vs. RCS pressure data from Table 1. The values in Table 4 are calculated at the minimum LTOPS temperature (i.e., the bolt-up temperature), which is slightly limiting for the MI transient.

The overshoot and undershoot pressures resulting from the HI events are shown in Table 5 as a function of setpoint pressure and RCS/SG temperature with 50°F temperature differential between the SG and RCS.

Table 4: Mass Injection Pressure Overshoots/Undershoots Summary

a,c

Table 5: D.C. Cook Unit 2 Heat Injection Pressure Overshoots/Undershoots Summary

a,c

a,c

a,c

5.2 PORV Setpoints Determination

Using the results of the LTOPS design basis MI and HI parametric transient analyses from Tables 4 and 5, the LTOPS maximum allowable PORV setpoints valid up to 48 EFPY are determined. LTOPS PORV setpoints for D.C. Cook Unit 2 are selected such that the LTOPS acceptance criteria, as defined in Section 4.0, are met. The maximum allowable PORV setpoint is determined based on the adjusted Appendix G limit or the PORV piping limit, whichever is more limiting. The adjusted Appendix G limit is the Appendix G limit minus the transmitter ΔP and wide range pressure instrument uncertainty (see Table 6).

A summary of the maximum allowable LTOPS PORV setpoint calculations and associated limits for the MI transient is shown in Table 7 and for the HI transient in Table 8. The maximum allowable PORV setpoints for the MI and HI transients are plotted as a function of indicated RCS temperature in Figure 1. The final maximum allowable PORV setpoint is determined such that it bounds both the MI and HI transient maximum allowable PORV setpoints. It should be noted that LTOPS PORV setpoints are shown up to 300°F, which is conservatively above the LTOPS enable temperature described in Section 5.4. Although a pressurizer steam bubble is not required by TS LCO 3.4.9 until Mode 3 (i.e., RCS temperatures ≥ 350°F), it is expected that a steam bubble will be required by plant operating procedures prior to disarming the LTOPS such that the PORV piping limit is not a concern and the pressurizer safety valves are available for overpressure protection per LCO 3.4.10 without being subjected to water relief.

Figure 1 also shows the current D.C. Cook LTOPS PORV setting of 435 psig. This indicates that the current LTOPS PORV setpoint will continue to protect the 48 EFPY isothermal P-T limits against the design basis HI transient across the full range of temperatures applicable to LTOP. However, the LTOPS PORV setpoint would need to be reduced drastically to accommodate the MI transient at indicated RCS temperatures < 200°F. The LTOPS setting necessary to provide this protection is 261 psig, which is judged to be impractical as there would be insufficient operating region to perform heatup and cooldown operations. Therefore, the RHR suction relief valve evaluation in Section 5.3 will demonstrate that the RHR suction relief valve is capable of providing protection against the MI transient such that the existing LTOPS PORV setpoint can be maintained.

Table 6: Adjusted Appendix G Limits for D.C. Cook Unit 2 for 48 EFPY

a,c

a,c

Table 7: D.C. Cook Unit 2 Maximum Allowable Setpoint Determination for the MI Transient

a,c

[ ] a,c

Table 8: D.C. Cook Unit 2 Maximum Allowable Setpoint Determination for the HI Transient

a,c

a,c

[ ]a,c

Figure 1: D.C. Cook Unit 2 Maximum Allowable LTOPS PORV Setpoint

(includes Pressure and Temperature Uncertainties)

5.3 RHR Suction Relief Valve LTOP Evaluation

As discussed in Sections 2.0 and 3.0, the RHR suction relief valve is a spring loaded water relief valve that is guaranteed to achieve full capacity at 110% of the nominal set pressure (i.e., 495 psig). The RHR suction relief valve is required to maintain the RHR system pressure below 110% of the design pressure (660 psig). Additionally, when credited as a pressure relief capability for LTOP, the RHR suction relief valve must maintain the pressure in the reactor vessel below the isothermal Appendix G P-T limit.

As long as the RHR suction relief valve capacity meets or exceeds the required relief capacity during the design basis LTOP transients, the pressure at the inlet to the relief valve will not exceed 495 psig. Therefore, the RHR suction relief valve evaluation primarily focused on determining if the RHR suction relief valve has sufficient capacity to relieve the design basis transients. When the relief valve has sufficient capacity, the peak pressure at the reactor vessel mid-plane will be the 495 psig accumulation pressure plus the applicable pressure drop back to the vessel and the peak pressure in the RHR system will be the 495 psig accumulation pressure plus the RHR pump head.

As calculated in Section 2.1, the maximum pressure drop from the reactor vessel midplane to the RHR suction relief valve inlet with four RCPs running and a conservative [ ]^{a,c} gpm relief flow is [ ]^{a,c} psi. Therefore, as long as the RHR suction relief valve capacity is not exceeded, the maximum pressure in the reactor vessel would be [ ]^{a,c} psig (495 psig accumulation pressure + [ ]^{a,c} psi ΔP). Since this peak pressure is below the lowest Appendix G P-T limit of 620 psig, the RHR suction relief valve will meet the LTOP acceptance criterion as long as the valve capacity is not exceeded by the design basis MI and HI transients.

An RHR pump head of [ ]^{b,c} psi was provided in Reference 4. Therefore, as long as the RHR suction relief valve capacity is not exceeded, the maximum pressure in RHR system would be [ ]^{a,c} psig (495 psig accumulation pressure + [ ]^{b,c} psi pump head). Since this peak pressure is below the RHR system pressure limit of 660 psig, the RHR suction relief valve will protect the RHR system as long as the valve capacity is not exceeded by the design basis MI and HI transients.

Based on the above, the RHR suction relief valve will protect both the reactor vessel Appendix G P-T limit and 110% of the RHR design pressure as long as the valve capacity is not exceeded during the design basis MI and HI transients. Table 2 summarizes the RHR suction relief valve capacity as a function of temperature from Reference 9, which was confirmed to remain valid in Reference 4.

The RHR suction relief valve capacity listed in Table 2 is evaluated for the design basis MI and HI transients as follows.

5.3.1 Mass Injection Transient

As discussed in Section 2.1, the design basis MI transient consists of the flow from two charging pumps injecting with letdown isolated. The design basis MI flow rate as a function of cold leg (or RCS) pressure is shown in Table 1. Interpolating the design basis MI flow rate at a conservatively low RCS pressure of 495 psig results in an injection flow of [ ]^{a,c} gpm. This is conservative because the actual RCS cold leg pressure would be higher at the RHR suction relief valve accumulation pressure (i.e., [ ]^{a,c} psig as discussed above), resulting in a lower injection flow rate. The injection flow rate of [ ]^{a,c} gpm is less than the RHR suction relief valve capacity with a 100 psig Pressurizer Relief Tank (PRT) backpressure across all temperatures at which the RHR system can be aligned (i.e., RCS temperatures < 350°F), which bounds the range of LTOP applicability (i.e., $60°F \leq T_{RCS} \leq 291°F$). Therefore, the RHR suction relief valve can protect both the isothermal reactor vessel Appendix G P-T limit and 110% of the RHR system design pressure against the design basis MI transient across all applicable conditions.

5.3.2 Design Basis Heat Injection Transient

Various cases of the design basis HI transient were analyzed to determine the required relief capacity of the RHR suction relief valve for RCS conditions across the range of LTOP applicability. Figure 2 illustrates the fluid expansion rate (or required relief rate) as a function of time for each of the cases analyzed. Table 9 summarizes the peak fluid expansion rates, which represent the required relief capacity, for each of the cases analyzed. Figure 3 plots the RHR relief capacity data from Table 2 along with the required relief capacity from Table 9. The RHR relief capacities are shown both with and without the PRT rupture disc intact. After the PRT rupture disc bursts, the RHR suction relief valve backpressure decreases and the effective relief capacity increases. Therefore, Figure 3 helps quantify the margin that exists between different backpressure assumptions.

As shown in Figure 3, the required relief capacity exceeds the installed RHR relief capacity for HI transients initiated at higher RCS temperatures. The installed relief capacity with a PRT backpressure of 100 psig is exceeded for HI transients initiated from a minimum indicated RCS temperature of 150°F. This temperature is increased to 166°F with credit for the installed relief capacity after the PRT rupture disc fails. Above these temperatures, the RHR suction relief valve is not capable of providing sole protection against the design basis HI transient. Therefore, two pressurizer PORVs will be required to be operable to above this temperature to provide protection against the design basis HI transient if no RCPs are running. As shown in Section 5.2, a single PORV with the current LTOPS PORV setting of 435 psig is capable of protecting the Appendix G P-T limits against the HI transient for the full range of temperatures applicable to LTOP.

For situations where the RHR system is aligned to the RCS at temperatures above 166°F, 110% of the RHR system design pressure will be protected from the design basis HI by the pressurizer PORVs (which are required to be operable for LTOP if no RCP is running) and/or a pressurizer steam bubble, working in conjunction with the RHR suction relief valve.

Table 9: Required RHR Suction Relief Valve Capacity for the HI Transient

a,c

a,c

Figure 2: Fluid Expansion Rate as a Function of Time

a,c

Figure 3: Installed RHR Relief Capacity and Required RHR Relief Capacity for the HI Transient as a Function of RCS Temperature

5.3.3 Analysis of HI Transients that can Occur with RCPs Running

As shown in Section 5.3.2, the RHR suction relief valve is unable to provide protection against the design basis HI transient initiated at indicated RCS temperatures greater than 150°F. However, the design basis HI transient can only occur from an initial condition in which no RCPs are running. If even a single RCP is running, sufficient flow will be maintained through each SG to keep the secondary side temperature coupled to the primary side temperature. Therefore, AEP may choose to credit this by recognizing that if at least one RCP is running, then the RHR suction relief valve is capable of providing LTOP over the full range of applicability. The pressurizer PORVs would then only be required to be operable if indicated RCS temperature is greater than 150°F (166°F with credit for PRT rupture disc) and no RCPs are running. These are the conditions in which the design basis HI transient can occur where the RHR suction relief valve cannot provide protection.

The LTOP LCO needs to ensure overpressure protection is provided for any overpressure transient that can occur under the allowed operating conditions. The design basis HI transient resulting from an RCP start is analyzed because it bounds all other heatup transients initiated at low temperatures. To support elimination of the RCP start HI transient if RCP(s) are running, less severe HI transients need to be evaluated to demonstrate that the RHR relief valve can provide protection.

As part of the original development of the LTOPS in Reference 13, the following HI transients were studied:

- Inadvertent actuation of the pressurizer heaters
- Loss of Decay Heat Removal (DHR)
- RCP start with the SG secondary side 50°F hotter than the primary side
- RCP start with cold charging and seal injection water accumulated in the pump suction leg

The two HI transients that result from an RCP start with temperature asymmetry were more severe than the other transients. These two transients can be removed from consideration if at least one RCP is running and Reference 13 showed that the loss of DHR transient is the next most severe HI transient. This section will analyze various cases of the loss of DHR and other HI transients to demonstrate that the RHR suction relief valve has adequate capacity to provide LTOP against all potential HI transients if at least one RCP is running.

Description of Analyses and Assumptions:

<u>*Loss of DHR*</u>

As described above and in Reference 13, the loss of Decay Heat Removal (DHR) is expected to be the most severe LTOP heatup transient that can occur from an initial condition with at least one RCP running. The following describes the conservative modeling of this transient as well as sensitivity cases that were analyzed to determine the required relief capacity.

a,c

[ ]a,c

The following sensitivity cases of the loss of DHR transient were analyzed:

[ ]a,c

Additional HI Transients

To revalidate the results of Reference 13 as well as provide quantified relief rates for additional types of HI transients, the following cases were analyzed:

- Plant heatup from continuous RCP heat from four pumps running ([ ]^{a,c} MWt) from a water solid initial condition. No other heat sources were modeled, and assumptions were similar to those described for the loss of DHR cases.
- Inadvertent pressurizer heater actuation ([ ]^{a,c} kW) with RCP heat from four pumps running. No other heat sources were modeled, and assumptions were similar to those described for the loss of DHR cases.

Results

The results of each additional HI transient are summarized in Table 10 and Figure 4. These results confirm that the loss of DHR transient bounds pressurizer heater actuation and RCP heat input transients. Figure 5 illustrates the relief rates for the loss of DHR cases as a function of RCS temperature. This demonstrates that for the same heat input rate, the relief rate is primarily a function of RCS temperature. Lower heat input rates extend the time to hot leg saturation and the peak relief at the time of saturation is lower. Therefore, analyzing the loss of DHR from 60°F through hot leg saturation bounds the relief capacity for a loss of DHR transient initiated at any temperature within the range of temperatures applicable to LTOP. Figure 6 shows the relief rates for the pressurizer heater actuation and RCP heat input transients.

The results show that the peak relief rates for each transient are well within the relief capabilities of the RHR suction relief valve. Therefore, if at least one RCP is running, the RHR suction relief valve is capable of providing protection against any potential HI transients. Section 5.3.1 showed that the RHR suction relief valve has sufficient capacity to protect against the design basis MI transient. Therefore, the RHR suction relief valve is capable of providing LTOP over the full temperature range of LTOP applicability if at least one RCP is running at indicated temperatures above 150°F (166°F with credit for PRT rupture disc). If all RCPs are stopped at temperatures greater than 150°F, the pressurizer PORVs are required to be operable to provide LTOP against the design basis HI transient.

Table 10: Results of the HI Transients that can Occur with at Least One RCP Running

a,c

a,c

Figure 4: Relief Rates as a Function of Time for Additional HI Transients

a,c

Figure 5: Relief Rates as a Function of RCS Tavg for Loss of DHR HI Transients

a,c

Figure 6: Relief Rates as a Function of Time for Bounded HI Transients

5.4 LTOPS Arming / Enable Temperature

The LTOPS enable temperature was calculated in Reference 11 using the methods of ASME Code Case N-641 to be [ ]^{a,c} °F (without uncertainty). With the temperature uncertainty of [ ]^{a,c} °F applied, the minimum arming/enable temperature is 291°F. The Technical Specifications LCO 3.4.10 and LCO 3.4.12 currently specify an LTOP enable temperature of 299°F, which remains conservative, if AEP desires to maintain it.

The current Unit 2 LTOP enable temperature of 299°F specified in the Technical Specifications bounds the enable temperature calculated for Unit 2 in this report and Unit 1 in Reference 14, and can be used at both units if desired, to maintain consistency.

5.5 Summary of Results and Conclusions

The design basis MI and HI transients were analyzed and used for LTOP evaluations of the pressurizer PORVs and RHR suction relief valve for the updated P-T limits to 48 Effective Full-Power Years (EFPY) as described in Sections 5.1 through 5.4. The following conclusions were drawn:

- A single pressurizer PORV (both PORVs are required to be operable with one assumed to fail), with the current maximum allowable LTOP pressurizer PORV setting of ≤ 435 psig, is capable of providing the following protection (Section 5.2 and Figure 1):
  - The design basis HI transient is protected over the full temperature range applicable to LTOP (i.e., $60 \leq T_{RCS} \leq 291$°F).
  - The design basis MI transient from two CCPs injecting is protected for indicated RCS temperatures ≥ 200°F. Below this temperature, alternate means of LTOP (i.e., an RCS vent or the RHR suction relief valve) are required to be operable.
  - The LTOPS PORV setpoint reduction necessary to protect against the MI transient at the lowest temperatures of LTOP applicability was judged to be impractical as there would be insufficient operating region to perform heatup and cooldown operations.
- The RHR suction relief valve, with the current nominal lift setting of 450 psig, is capable of providing the following protection (Section 5.3):
  - The design basis MI transient from two CCPs injecting is protected over the full temperature range where the RHR system can be aligned (i.e., $T_{RCS} \leq 350$°F), which bounds the applicable LTOP temperature range (i.e., $60 \leq T_{RCS} \leq 291$°F).
  - The design basis HI transient is protected for indicated RCS temperatures ≤ 150°F (≤ 166°F with credit for PRT rupture disc failure). Above this temperature, two pressurizer PORVs are required to be operable to provide this protection.
  - For situations where the RHR system is aligned to the RCS at indicated temperatures above 166°F, 110% of the RHR system design pressure will be protected during a design basis HI transient by the pressurizer PORVs (which are required to be operable for LTOP) and/or a pressurizer steam bubble, working in conjunction with the RHR suction relief valve.
  - With at least one RCP running, sufficient flow and heat transfer is maintained through each SG such that the conditions associated with the design basis HI transient cannot occur. The analyses in Section 5.3.3 demonstrated that the RHR suction relief valve is capable of providing protection against the remaining HI transients that can occur with at least one

RCP running over the full temperature range where the RHR system can be aligned (i.e., $T_{RCS} \leq 350°F$).

- When the RCS is depressurized, LTOP can be provided by an RCS vent of ≥ 2.0 square inches or any single pressurizer PORV blocked open. This includes protection against the design basis MI transient resulting from two CCPs injecting.
- The analysis does not credit or impose any limitations on the maximum number of RCPs allowed to be in operation throughout the range of LTOP applicability.
- Accumulators must be isolated or maintained at a pressure less than the maximum RCS pressure allowed by the P-T limit curves for the existing RCS cold leg temperature.
- The minimum LTOPS arming/enable temperature (including temperature uncertainty) is 291°F (Section 5.4).

Based on the above conclusions, the LTOP acceptance criterion to protect the isothermal Appendix G P-T limits is met with the following minimum relief capabilities required to be operable (See Figures 7 and 8):

- For $60 \leq T_{RCS} \leq 150°F$[1] with zero through four RCPs running:
  - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the MI and HI transients.
- For 150[1] < T_{RCS} < 200°F:
  - With zero RCPs running:
    - The RHR suction relief valve, with a setpoint of ≤ 450 psig, is required to be operable and will protect against the MI transient; and
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against the HI transient.
  - With at least one RCP running:
    - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the MI and HI transients.
- For $200 \leq T_{RCS} \leq 291°F$:
  - With zero RCPs running:
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against both the MI and HI transients.
  - With at least one RCP running:
    - The RHR suction relief valve, with a setpoint ≤ 450 psig, is required to be operable and will protect against both the MI and HI transients; or
    - Two pressurizer PORVs, with lift settings ≤ 435 psig, are required to be operable and will protect against both the MI and HI transients.

The minimum LTOP requirements calculated for D.C. Cook Unit 1 in Reference 14 bound those described above and can be used for D.C. Cook Unit 2, if desired, to maintain consistency between both D.C. Cook units.

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[1] ≤ 166°F with credit for Pressurizer Relief Tank (PRT) rupture disc failure

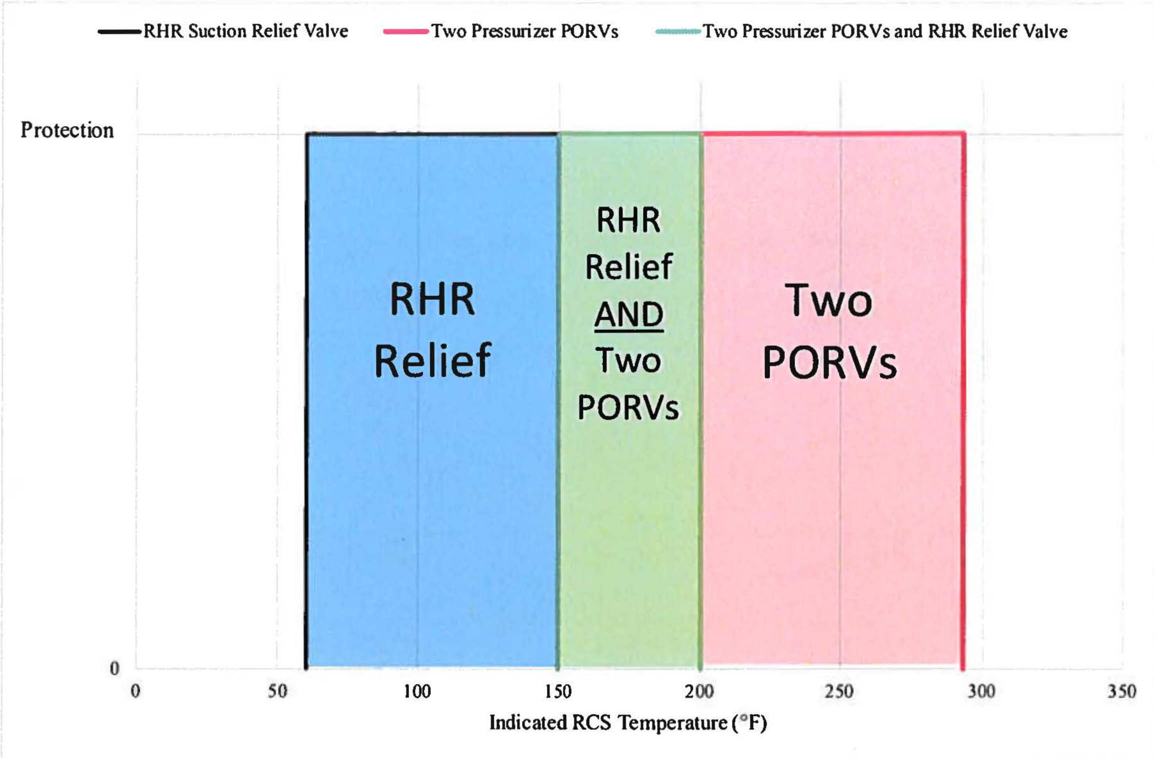

Figure 7: D.C. Cook Unit 2 Minimum Required Relief Capabilities for LTOP with Zero through Four RCPs Running

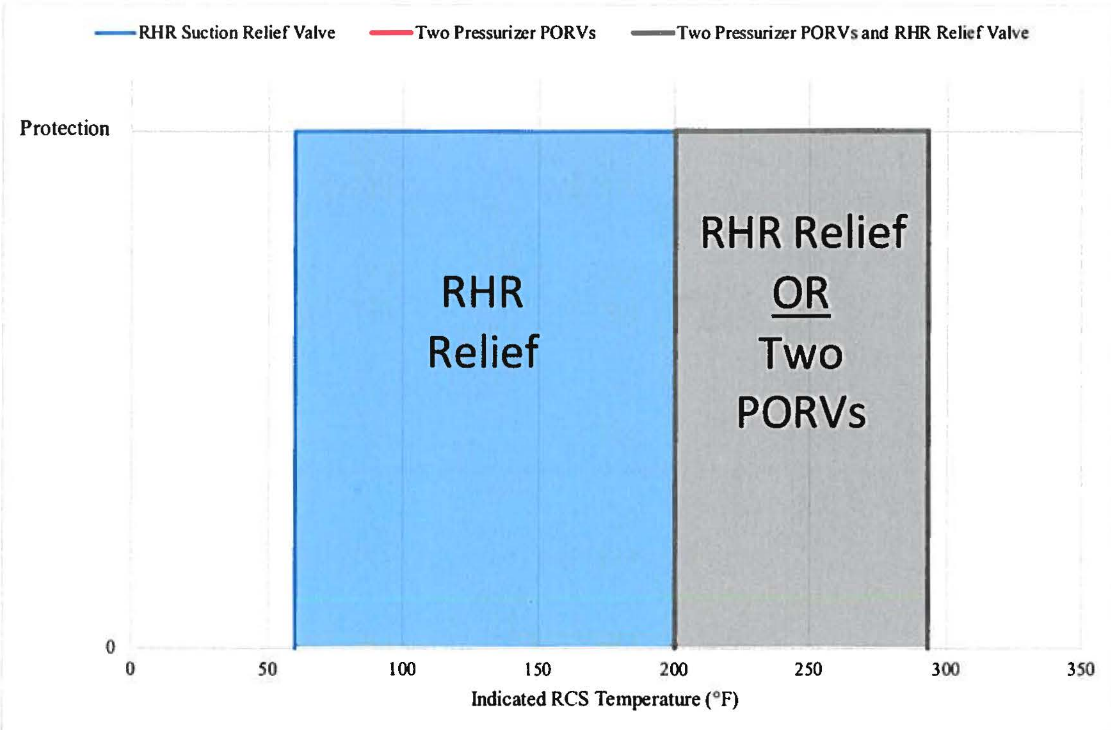

Figure 8: D.C. Cook Unit 2 Minimum Required Relief Capabilities for LTOP with at Least one RCP Running

6.0 References

1. Westinghouse Report WCAP-18456-NP, Rev. 0, "D.C. Cook Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," February 2020.

2. a,c

3.

4.

5. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

6. a,c

7. Westinghouse Report WCAP-11640, Revision 0, "Cold Overpressure Mitigation System Deletion Report," March 1988.

8. Westinghouse Report WCAP-11736-A, Volumes I & II, Revision 0.0, "Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners Group," October 1989.

9. Westinghouse Report WCAP-13235, Revision 0, "Donald C. Cook Units 1&2 Analysis of Low Temperature Overpressurization Mass Injection Events with Pressurizer Steam Bubble and RHR Relief Valve," March 1992.

10. a,c

11.

12. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800, BTP 5-2, Revision 3, "Overpressurization Protection of Pressurized-Water Reactors while Operating at Low Temperatures," March 2007. (ML070850008)

13. Westinghouse Report WCAP-10529, Revision 1, "Cold Overpressure Mitigating System," November 1985.

14. Westinghouse Letter LTR-SCS-19-50, Revision 0, "D.C. Cook Unit 1 Low Temperature Overpressure Protection System (LTOPS) Analysis for 48 EFPY," March 2020.

15. Westinghouse Report WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

Approval Information
Author Approval Mitchell Luke J Jun-30-2020 11:20:22
Author Approval Joseph Thomas G Jun-30-2020 11:36:38
Verifier Approval Jaskiewicz Bryan D Jun-30-2020 13:06:07
Manager Approval Billman Steven Jun-30-2020 13:24:49

Files approved on Jun-30-2020

Enclosure 8 to AEP-NRC-2021-28

Affidavit of Withholding Pursuant to 10 CFR 2.390, Westinghouse Electric Company

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Korey L. Hosack, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of LTR-SCS-20-18-P be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These

AFFIDAVIT

lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2020 06 29

Korey L. Hosack, Manager
Licensing, Analysis, & Testing