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Kewaunee Nuclear Power Plant
Operated by Nuclear Management Company, LLC

NRC-03-016 10 CFR 50 90

February 27, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

KEWAUNEE NUCLEAR POWER PLANT
DOCKET 50-305
LICENSE No. DPR-43
NMC RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION CONCERNING LICENSE AMENDMENT REQUEST 187 TO THE KEWAUNEE NUCLEAR POWER PLANT TECHNICAL SPECIFICATIONS (TAC NO. MB5718)

References:

1) Letter from Mark E. Warner (NMC) to Document Control Desk (NRC), "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse Vantage + Fuel," dated July 26, 2002.

2) Letter from John G. Lamb (NRC) to Thomas Coutu (NMC), "Kewaunee Nuclear Power Plant – Request for Additional Information Regarding Proposed Amendment Request Conforming Technical Specification Changes for Use of Westinghouse Vantage + Fuel," (TAC NO. MB5718) dated January 21, 2003

The Nuclear Management Company, LLC, (NMC) is submitting this response to the Nuclear Regulatory Commission (NRC) request for additional information (RAI) (reference 2) concerning License Amendment Request (LAR) 187 (reference 1) to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) revising KNPP TS to allow transitioning to Westinghouse 422V+ nuclear fuel.

Attachment A to this letter contains the NRC RAI questions with NMC responses. Attachment B contains the non-LOCA events analysis from the reload transition safety report supplied by Westinghouse to the NMC for transitioning to the 422V+ fuel Attachment C contains a matrix of Westinghouse standard technical specifications to the KNPP TS submitted in reference 1. Attachment D contains a retraction of the loss of normal feedwater (LONF) event. Attachment E contains strike-out Technical Specification pages which are being modified due to the NRC RAI and submitted herein Attachment F contains the affected Technical Specification pages as revised. Attachment G contains LAR 187a KNPP TS Basis strike-out pages Attachment H contains LAR 187a KNPP TS Basis revised pages. Attachment I contains LAR 187a revised COLR pages. Finally, Attachment J contains a list of the commitments the NMC is making associated with this response.

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Attachment A contains NMC responses to NRC questions associated with the NMC's submittal requesting approval to transition KNPP's nuclear fuel to Westinghouse 422V+ fuel. To avoid confusion with the attachments, this letter's attachments have been labeled alphabetically, whereas the attachments in the reference 1 letter are numerically labeled. Therefore, if a reference is stated as a numerical attachment, that attachment is contained in the reference 1 letter. If an alphabetical attachment is referenced, then that attachment is contained as a part of this letter.

In Attachment A the questions and responses are grouped into four categories. These categories refer to attachments contained in the reference 1 letter. The categories are:

1. Attachment 1 Questions – Safety Analysis
2. Attachment 2 Questions – Technical Specifications
3. Attachment 3 Questions – Westinghouse Report
4. WCAP-15591 Questions – Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology

Although category 3 questions are labeled as being associated with attachment 3, they relate to reference 1, attachment 4. Reference 1, Attachment 3, and also reference 1, Attachment 2, relate to proposed changes in KNPP Technical Specifications. Attachment 4 is the Westinghouse transition report the questions refer. When changing categories, the question numbers restart at number one.

Included with some of the NMC's responses are additional references. These references are included to supplement the response given. The NMC's goal is to respond to the questions with sufficient detail without providing unnecessary information. The references provide an additional source(s) of details if additional information is needed.

Attachment B contains section 5.1, "Non-LOCA Transients," of the Reload Transition Safety Report (RTSR) that Westinghouse prepared for the NMC to provide the information required for transitioning to Westinghouse 422V+ fuel. From the RTSR, NMC and Westinghouse condensed the information to that considered necessary for the NRC to complete their review to approve the use of 422V+ fuel in the KNPP reactor. As this section from the RTSR helps to answer numerous NRC staff questions, it is included.

Attachment C contains a matrix comparing NUREG 1431, Revision 2, "Westinghouse Standard Technical Specification (STS)," to the KNPP Technical Specifications, submitted in reference 1. Reference 2, attachment 2 question 3a requested this comparison. The shaded cells of the matrix highlight differences between STS and the submitted KNPP TS. Text in the shaded areas of the matrix states the requirements, as stated in the reference 1 submittal, and the proposed changes, in parentheses, to conform to STS.

Attachment D contains a retraction of the loss of normal feedwater (LONF) accident analysis included as part of the reference 1 submittal. On review of this accident analysis, the NMC concluded that the submitted LONF analysis contained actions that the NMC would like to further review prior to submittal for NRC approval. As this reanalysis is only necessary to support a Stretch Power Uprate for the KNPP, the NMC is retracting the LONF analysis contained in reference 1 and will resubmit the analysis when requesting a Stretch Power Uprate.
Attachments E, F, G, H, and I contain changes to KNPP TS, TS Basis, and KNPP core operating limits report These pages supersede those pages submitted in reference 1 and are marked as LAR 187a. Only those pages changed due to this response to the NRC questions are submitted as part of this letter.

Attachment J contains those commitments the NMC is making to support approval of the transition of KNPP fuel to Westinghouse 422V+ fuel. Any response to an NRC question which appears to contain a commitment, but that statement is not listed in Attachment J, is not to be considered a commitment made by this submittal by the NMC.

In reference 1, the NMC requested approval of LAR 187 by February 2003. Due to the period required responding to these RAI questions the NMC is hereby revising this approval date request. The NMC is starting the KNPP 2003 refueling outage on April 5, 2003. To allow for loading of Westinghouse 422V+ fuel during this outage NMC requests approval by April 1, 2003.

The purpose of this submittal is to provide additional information to the NRC staff for their review of KNPP LAR 187 and does not significantly change the analysis section of the NMC's original submittal. Therefore, the safety analysis, no significant hazards determination, and environmental considerations included in reference 1 are still applicable.

In reference 2, the NRC requested a response within 30 days of the date of the letter. As stated in this letter, if a change in this response date was needed, NMC should contact the NRC Project Manager for KNPP. On February 20, 2003, the NRC Project Manager was contacted and an extension was granted until February 28, 2003.

I declare under penalty that the foregoing is true and correct.
Executed on February 27, 2003.

Thomas Coutu

Thomas Coutu
Site Vice-President, Kewaunee Plant

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Attachments·

A. NRC RAI Questions with NMC responses
B. Westinghouse Reload Transition Safety Report section 5.1, "Non-LOCA Transients"
C Westinghouse Standard Technical Specification to KNPP LAR 187 TS Matrix
D. Loss of Normal Feedwater Event Retraction
E. LAR 187a KNPP TS Strike-out Pages
F. LAR 187a KNPP TS revised pages
G. LAR 187a KNPP TS Basis strike-out pages
H. LAR 187a KNPP TS Basis revised pages
I. LAR 187a revised COLR Pages
J. Commitments

cc- US NRC, Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

ATTACHMENT A

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

KEWAUNEE NUCLEAR POWER PLANT
REQUEST FOR ADDITIONAL INFORMATION
REGARDING TECHNICAL SPECIFICATION CHANGES
FOR USE OF WESTINGHOUSE VANTAGE PLUS FUEL
(TAC NOS MB5718)

Attachment 1 Questions
Safety Analysis

1. Page 3 of Attachment 1 lists a number of proposed Technical Specification (TS) changes associated with this license amendment request However, the marked up TS pages are not included in the package. Please provide the marked up TS pages

NMC Response: Although page 3 of Attachment 1 does list a number of proposed Technical Specification (TS) changes associated with this license amendment request, the actual changes are to the COLR document and are not changes to the TS pages themselves. Page 3 of attachment 1 lists TS sections 2.6, 2 7, 2 9, 2.10, 2 11, 2.12, and the associated figures - these are the sections in the COLR and the figures in the COLR.

2. Page 4 of Attachment 1 refers to References 1-17, 1-18 and 1-19, and Reference 5, Table 5.1-8 We are unable to locate these references in the amendment package Please provide us with clearer information regarding these references

NMC Response: For the transition to Westinghouse V422+ fuel Westinghouse developed a reload transition safety report (RTSR) for NMC. Included in the RTSR, as Appendix F, is a licensing summary of the entire RTSR report. This section was designed to contain all the necessary information for the Nuclear Regulatory Commission (NRC) staff to approve the KNPP fuel transition in a non-proprietary document This non-proprietary report was transmitted to the NRC as Attachment 4 to the KNPP license amendment request (LAR) 187.[1] and was entitled "Westinghouse Report, Technical Design Basis for the Transition to 422V+ Fuel."

During the LAR submittal development, reference to these documents was included in Attachment 1 with the assumption that these references would be included in the non-proprietary report to be sent to the NRC Subsequently, these references were excluded from the report but not removed from Attachment 1 of the submittal These documents are:

- Reference 1-17, "Letter from D.J. Wanner (NMC) to A Meliksetian (W), 'Kewaunee Input Assumptions – Initial Table Revisions,' September 29, 2000."
- Reference 1-18, "Letter from D.J. Wanner (NMC) to R.H. Owoc (W), 'Kewaunee Risk Input Assumptions and LOCA Input Confirmation,' February 15, 2002 Transmittal."
- Reference 1-19, "Letter from D.J. Wanner (NMC) to R.H. Owoc (W), 'Formal Transmittal of Kewaunee Fuel Storage Rack Information,' April 22, 2002."
- Reference 5, Table 5.1-8, "Transients Evaluated or Analyzed for the KNPP Fuel Upgrade/Power Uprating Program "

(1) Letter from Mark E Warner (NMC) to Document Control Deck (NRC), "License Amendment Request 187 to the Kewaunee Nuclear Power Plant Technical Specifications, Conforming Technical Specification Changes for Use of Westinghouse VANTAGE + fuel," dated July 26, 2002

The first three references document the transmittal of safety analysis input assumptions; the final reference lists the non-LOCA safety analyses that are within the scope of the RTSR and correlates them to the associated USAR section. The last reference will be included in Attachment B.

References 1-17, 1-18 and 1-19 have been excluded from the report and are for internal use between NMC and their vendor. These inputs are used in the development of safety analyses in support of the fuel transition to Westinghouse fuel. However, LAR 187, Attachment 4, Section 1.5 (Performance Capabilities Working Group Parameters), Table 1-2, provides a summary sheet of several primary parameters used by all analysis groups to ensure consistent use of inputs

3. Page 5 of Attachment 1 states that, "Empirical data acquired during Cycle 25 confirms that this fuel is both compatible with KNPP reactor design and with the Framatome/ANP fuel currently in use." Please discuss the data acquired and technical basis for reaching this conclusion

NMC Response: Four (4) Westinghouse assemblies of the 422V+ fuel design (assemblies E87 – E90) were introduced into Cycle 25 under the KNPP Lead Test Assembly (LTA) Program (Reference 1 and 4). The KNPP LTA program requires the evaluation of lead test assemblies for mechanical compatibility considering nozzle, spacer, fuel rod, and fuel pellet design changes with respect to the co-resident fuel, impact on cycle length and ability to operate at full power, ability to model the LTAs with neutronic and safety analysis methods; validity of current vendor analyses such as LOCA and fuel rack criticality, thermal and radiological analyses; ability to store LTAs in the new fuel pit and/or spent fuel pool, reduction of margin, including potential restrictions of $F_{\Delta H}$ and F_Q in LTAs or non-LTAs, interaction of LTAs with co-resident fuel, including the effect of multiple changes to multiple LTA types; discharge burnup limit requirements, including corrosion effects and rod bow, fuel handling concerns; impact of different burnable absorber designs, if applicable; ramp rate restrictions; reactor coolant system chemistry issues; fuel reliability issues; setpoint changes, ALARA issues; reactivity management implications, and operation impacts related to MTC, Axial offset, etc. The applicable issues are addressed in the Westinghouse LTA fuel rod design report (Reference 2).

In addition, the LTA program requires the monitoring of and comparison to prediction for the following parameters: core-wide startup physics test parameters; flux map parameters of core-wide and LTA-specific $F_{\Delta H}$, core-wide and LTA-specific F_Q; peak LTA-specific fuel rod burnups, and LTA-specific axial offset for axially varying LTAs

The LTAs have been successfully placed in KNPP Cycle 25 without incident. This confirms that the LTAs are mechanically compatible with the KNPP fuel handling equipment, reactor vessel and co-resident Framatome-ANP fuel All measured and inferred startup physics test parameters (Low Power Physics Test Parameters) for Cycle 25 have met the acceptance criteria relative to the prediction for Cycle 25 (Reference 3) This confirms that beginning of cycle mixed-core effects are correctly predicted and that no anomalous compatibility issues exist. Flux maps 2501 through 2519 have been processed and the $F_{\Delta H}$ and F_Q for the LTA and non-LTA fuel meet the applicable limits and have trended as predicted Peak LTA rod burnups compare acceptably with prediction

The LTAs are not axially varying and therefore the LTA-specific axial offset is not required to be compared. However, the core-wide axial offset is trending acceptably with prediction. This data confirms that the mid-cycle mixed-core effects are correctly predicted to date and that no anomalous compatibility issues exist to date. The KNPP LTA program requires the documentation of the LTA comparisons in a report that is issued after the completion of Cycle 25.

References

1. WPSC Letter NRC 91-084 from K H. Evers to U.S. Nuclear Regulatory Commission Document Control Desk, entitled "Core Reloads of Advanced Design Fuel Assemblies", dated June 19, 1991.
2. Westinghouse Report, entitled "Kewaunee LUA Fuel Rod Design Work Report, Fuel Performance Analysis Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assemblies", CAD-01-212, dated August 28, 2001.
3. NMC Report, entitled "Kewaunee Nuclear Power Plant, Cycle 25 Startup Report", dated February 2002.
4. Letter from John Lamb (NRC) to MR. Mark Reddemann (NMC), "Kewaunee Nuclear Power Plant – Issuance of Amendment (TAC NO. MB2205), dated August 13, 2001 (Lead Test Assemblies)

Attachment 2 Questions
KNPP Technical Specifications

1. The licensee is proposing to revise the TS Bases Section 2 1, "Safety Limits - Reactor Core," to include three DNBR correlations used in the safety analyses These three correlations are WRB-1, HTP and W-3. This is not consistent with TS 2.1.b, which provides DNBR limits for only WRB-1 and HTP DNB correlations Why is the W-3 correlation and its associated limit not specified in the proposed revision to TS 2.1? Why isn't the statistical methodology (RTDP) incorporated into the DNBR safety limit or discussed in the proposed TS Bases section?

NMC Response: TS 2 1 concerns the limiting combinations of thermal power, Reactor Coolant System (RCS) pressure and coolant temperature that are allowed during operation and hot stand-by modes The limits are presented in Figure 1 of the Core Operating Limits Report (COLR) in the form of safety limit curves. Only those Departure from Nucleate Boiling (DNB) correlations used to generate or validate the safety limit curves should be mentioned in TS 2 1

The HTP and WRB-1 DNB correlations are used in the generation and validation of the safety limit curves Validation of the safety limit curves is demonstrated, in part, by the protection offered by the ΔT trips, which are in turn validated by the safety analyses that rely on the ΔT trips. The W-3 DNB correlation is not used in the generation or validation of the safety limit curves. Therefore, only the WRB-1 and HTP DNB correlations should be mentioned in TS 2.1.

The discussion in the basis for TS 2.1 should include all DNB correlations used in the generation and validation of the safety limit curves, namely the HTP and WRB-1 DNB correlations. The associated correlation limits represent the inherent statistical reliability of the correlation predictions as compared to experimental measurement. The Revised Thermal Design Procedure (RTDP) (Attachment 6 to LAR 187) accounts for the additional statistical variability of the correlation predictions due to uncertainties in plant operating parameters (pressure, temperature, power, flow), nuclear and thermal parameters, fuel fabrication parameters and computer codes and statistically combines these uncertainties with the inherent DNB correlation uncertainties to determine the RTDP DNB ratio (DNBR) design limits. The RTDP DNBR design limits represent the increase in the correlation limit that is necessary to accommodate the additional statistical variability due to the factors mentioned while utilizing nominal plant operating parameters in the DNBR calculation.

It is not appropriate to include the RTDP DNBR design limits in the TS or TS bases or to discuss the details of the RTDP methodology in the TS bases. The RTDP DNBR design limits and RTDP methodology represent more detail than is typically included in the TS bases and are not necessary for the explanation of TS 2.1.

Since the adjustments to the correlation limits are direct properties of the specific instrument uncertainties and the methods used to evaluate these uncertainties, the RTDP DNBR design limits are not a property of the DNB correlation per se. As such, the RTDP DNBR design limits and the RTDP methodology used to derive them represent significantly more detail than is typically included in TS (e g , pump curve assumptions, fan coil unit performance, etc., are usually not included in the TS, but are documented in the Updated Safety Analysis Report).

Furthermore, the RTDP DNBR design limits are applicable only to event analyses performed with the RTDP methodology and therefore do not apply to all events analyzed, whereas the correlation limits are independent of the event under consideration. Therefore, including RTDP DNBR design limits and RTDP methodology would cause an unnecessary increase in the complexity of the TS bases Finally, inherent in the application of the RTDP methodology to a given event is the use of the RTDP DNBR design limits to verify that no departure from nucleate boiling occurs during the event By applying the RTDP methodology and using the RTDP DNBR design limits, it is verified that the *correlation limit* is not violated. In other words, end result of the DNB analysis is to show that the *correlation limit* is not violated. The statistical methodology (RTDP) accomplishes this by employing the RTDP methodology and the RTDP DNBR design limits and the non-statistical methodology applies the Standard Thermal Design Procedures and uses the correlation limits directly. In summary, the Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. With the STDP procedure, the nominal plant operating parameters with uncertainties applied are used to calculate the DNBR The DNBR limits for the STDP procedure are the appropriate correlation limits increased by sufficient margin to offset the applicable DNBR penalties. With the RTDP procedure, the nominal plant operating parameters are used to calculate the DNBR The DNBR limits for RTDP procedure are the appropriate RTDP design limits increased by sufficient margin to offset the applicable DNBR penalties

The TS bases will be revised to discuss only the DNBR correlations used to generate and validate the safety limit curves. The text of the last paragraph of the section entitled "Basis – Safety Limits – Reactor Core (TS 2.1)" is proposed as:

"Two departure form nuclear boiling ratio (DNBR) correlations <u>are</u> used in the generation and validation of the safety limit curves· the WRB-1 DNBR correlation and the high thermal performance (HTP) DNBR correlation. The WRB-1 correlation applies to Westinghouse 422V+ fuel The HTP correlation applies to Framatome ANP fuel with HTP spacers The DNBR correlations have been qualified and approved for application to Kewaunee. The DNB correlation limits are. 1.17 for the WRB-1 DNBR correlation and 1 14 for the HTP DNBR correlation "

References:
1. WCAP 11397-P-A, "Revised Thermal Design Procedure," April 1989

2 TS Sections 2.3.a 3 A and 2.3 a 3.B specify the Overtemperature and Overpower ΔT trip setpoint equations The proposed changes include a change in the definition of the variable T. The current KNPP TS define T as the Average temperature. The proposed TS change defines T as the Reference Average Temperature at Rated Power. Please provide the technical basis for this change. This change is not consistent with WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," or the Westinghouse Standard Technical Specifications.

NMC Response: The proposed change was erroneously applied to the definition of T. The definition was intended as a semantic change to the definition of T' and was for clarification purposes only. On review of these TS an additional change will be made to the definition of ΔT_0 The definition stated that the indicated ΔT_0 was in % Rated Power whereas the actual indication is in % The following definitions in TS 2 3.a 3 A and TS 2.3.a.3.B will be revised as indicated:

ΔT_0	=	Indicated ΔT at RATED POWER, %
T	=	Average Temperature, °F

Corresponding appropriate changes to the Technical Requirements Manual will also be made.

3. The licensee has proposed many changes to TS Section 3 10 - Control Rod and Power Distribution Limits. The licensee seems to be adopting portions of NUREG-1431, "Standard Technical Specifications (STS) Westinghouse Plants" while maintaining portions of the existing KNPP TS With respect to the proposed changes to TS Section 3.10

   a. Please provide technical justification for the deviation from STS action completion times

NMC Response: NUREG 1431, "Standard Technical Specifications – Westinghouse Plants," Revision 2 (STS), section 3.2, "Power Distribution Limits," and section 3 4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits," contain those technical specifications similar to the KNPP TS section 3.10. NMC is changing the methodology for core axial offset control from constant axial offset control (CAOC) to relaxed axial offset control (RAOC) As such, the following STS sections, using RAOC specifications, are applicable

1. 3.2.1B, "Heat Flux Hot Channel Factor ($F_Q(Z)$ (RAOC-W(Z) Methodology
2. 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)
3. 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)."
4. 3.2.4, "Quadrant Power Tilt Ratio (QPTR)
5. 3 4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits

Attachment C contains a matrix of the KNPP TS compared to NUREG 1431, Rev 2, "Standard Technical Specifications - Westinghouse Plants." This matrix lists the Standard Technical Specification (STS) actions if an LCO is exceeded and the associated completion time Next to the STS actions and completion times are listed the KNPP proposed actions and proposed completion times if an LCO is exceeded Those areas where there is a discrepancy between the STS and the KNPP TS are highlighted The discrepancies in actions and completion times found are:

1. STS has hot channel factor (HCF) actions and completion times for the situation where required actions and associated completion times are not met, KNPP TS does not. Except for $F_{\Delta H}^{N}$, which requires two hours to reduce reactor thermal power (RTP) to less than 5% versus the STS 6 hours.
2. KNPP TS has an allowance of 15 minutes when axial flux difference (AFD) exceeds its limits to restore AFD to within its limits whereas STS does not
3. KNPP TS has AFD actions that require reducing the power range nuclear instruments high flux reactor trip setpoint to 55% or less within 72 hours, STS has no such requirement.
4. KNPP TS has an AFD requirement to verify AFD within limits if the associated alarms are out-of-service (OOS), STS does not.
5. KNPP TS has a requirement to log quadrant power tilt when the quadrant power tilt monitor is OOS, STS does not.

For item 1 above, KNPP TS require that if the HCF's are not within limits greater than the allowed period KNPP TS 3 0 C, "Standard Shutdown Sequence," would be followed, except for $F_{\Delta H}^{N}$ which requires a reduction in RTP to less than 5% in 2 hours. To clarify the action to be taken for exceeding the HCF limits NMC will modify the $F_{\Delta H}^{N}$ completion time and add an action that is consistent with STS to require the KNPP RTP to be reduced to less than 5% within 6 hours for the $F_Q^N(Z)$ nd $F_Q^{EQ}(Z)$ limits

For item 2 above, this allowance is part of KNPP's current TS. NMC reviewed this allowance and concluded this allowance is associated with constant axial offset control (CAOC) and not with RAOC For consistency with RAOC, the 15-minute allowance to restore AFD to within limits will be removed from the KNPP TS

Item 3, above is a restriction that the current KNPP TS contains but STS does not. The STS basis for the actions required for AFD outside its limits states that as an alternative to restoring the AFD to within its specified limits a reduction in thermal power to < 50% RTP is acceptable. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. As a reduction in the NI high flux setpoint is not required to maintain the core in a safe condition and not included in STS, NMC will remove this requirement

Item 4 above is a restriction that the current KNPP TS contains but STS does not. As STS does not contain this restriction and the removal of this alarm from STS was approved by TSTF 110, Revision 2, NMC requests to remove this requirement from KNPP TS As part of this removal NMC will add the surveillance requirement contained in TSTF 110 Revision 2, to verify AFD within limits weekly

Item 5 above is a restriction that the current KNPP TS contains but STS does not. As these requirements are a part of KNPP's current TS NMC decided to maintain these requirements in our TS. They are a part of our current licensing basis and are more restrictive than STS

b STS for $F_Q^N(z)$ and $F_{\Delta H}^N(z)$ contain an action to eventually reduce to MODE 2 (≤5% RTP) if required actions and associated completion times cannot be satisfied. The proposed KNPP TSs does not include such an action statement Please provide justification for this

NMC Response NMC has reviewed this action statement and will include it in KNPP's TS. (see Attachment C)

c. The proposed KNPP TS changes incorporate a change in methodology from Constant Axial Offset Control (CAOC) to Relaxed Axial Offset Control (RAOC) The KNPP submittal does not discuss this change in methodology. Please discuss the purpose and basis for this change in methodology, provide a reference to the NRC approved methodology being applied and discuss the validity of this methodology for KNPP?

NMC Response: The Axial Flux Difference (AFD) is a measure of axial power distribution skewing to the top or bottom half of the core. It is very sensitive to core-related parameters such as control bank position, core power level, axial burnup, and axial xenon distribution The limits on AFD assure that the Heat Flux Hot Channel Factor $F_Q(z)$ is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD limits are used in the nuclear design process and assumed in the safety analyses as a boundary of possible initial condition axial power shapes Operation outside these limits during Condition I operation influences the possible power shapes and result in Condition II transients. Condition II transients, assumed to begin from within the AFD limits, are used to confirm the adequacy of Overpower Delta-T (OPΔT) and Overtemperature Delta-T Trip (OTΔT) Setpoints.

The CAOC methodology is presently incorporated into the Kewaunee Technical Specifications Section 3.10 Axial Flux Difference (AFD) Application of the RAOC and F_Q Surveillance Methodologies requires the alteration of the AFD and $F_Q(z)$ Technical Specifications Changes to Technical Specification 6.9 4 CORE OPERATING LIMITS REPORT (COLR) is also required to accommodate the methodology change

The current Technical Specification 3 10, per the COLR, specifies a target band of +5%, -5% for normal operation in the Operating Mode above 50% Rated Thermal Power (RTP) The RAOC Methodology typically allows an AFD operating space relaxation to –12%, +8% delta flux (ΔI) at 100% RTP and linearly increasing to –30%, +28% ΔI at 50% RTP, in the Operating Mode above 50% RTP Penalty minutes are not applicable to the RAOC Methodology and therefore reference to them is no longer required Reducing the power level to less than or equal to 50% RTP, this places the core in a condition for which the value of the AFD is not important in the applicable safety analyses In addition, a rapid rise in power to greater than 50 percent RTP with AFD outside limits does not immediately create an unacceptable situation. Since the transient analysis setpoint calculations for f(ΔI) (input to the overtemperature delta-temperature (OTΔT) trip function) are based on the same core power distributions that the fuel designers use for a reload cycle design, the OTΔT trip function provides an acceptable level of protection for such an excursion

Bases 3.10 is revised to be consistent with the RAOC and F_Q Surveillance Methodologies. The discussion concerning target flux difference is replaced by verbiage of monitoring the AFD consistent with the RAOC Methodology.

Specification 6 9 4 lists the approved analysis methodologies used for determining the cycle specific core operating limits specified in the COLR. The proposed change adds WCAP-10216-P-A, Rev. 1A since the RAOC and $F_Q(z)$ Surveillance Methodologies are replacing the Constant Axial Offset Control (CAOC) strategy as the approved methodology for generating the AFD and $F_Q(z)$ core operating limits

d Proposed TS 3.10 b 5 requires that the measured $F_Q^{EQ}(z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core Considering the new V422+ fuel and uprated power conditions, does the 80% value remain adequate?

NMC Response: The requirement that the measured F_Q^{EQ} (Z) hot channel factor under equilibrium conditions shall satisfy the relationship in the central axial 80% of the core is not a fuel-type related requirement. For NUREG-1431 Rev. 2, NUREG-0452 Rev. 5, and plants that have implemented these standards and the Relaxed Axial Offset Control Methodology, this requirement is only applicable to the central axial 70% of the core. The requirement for Kewaunee is more conservative than the standard requirement. As provided in NUREG-1431 Rev. 2, the top and bottom portions of the core are excluded from the surveillance because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions For the Westinghouse 422 V+ fuel, the reduced enrichment axial blankets of the assemblies will result in even less power being produced in the top and bottom 10% of the core relative to past cycles. Therefore, the requirement to monitor only the central axial 80% of the core for F_Q^{EQ} (Z) remains valid and will continue to ensure that acceptable power distributions are maintained in transition and equilibrium cores containing Westinghouse 422 V+ fuel

e. Proposed TS 3 10 b 6 B has a typo, the last portions should read as "in accordance with TS 3 10.b 6.A."

NMC Response: The change noted above will be made

f. Proposed TS 3 10.b.6 C refers to a peak pin power parameter of interest as F_Q^N. The current KNPP TS, which is similarly worded, refers to the peak pin power parameter as $F_{\Delta H}^N$. The proposed TS wording is identical to the current TS except that this parameter has changed from F_Q^N to $F_{\Delta H}^N$. Please provide the technical basis for this proposed change.

NMC Response TS 3 10 b.6.C will be revised to be consistent with NUREG-1431 Rev 2 and the Relaxed Axial Offset Control (RAOC) Methodology.

Implementing NUREG-1431 Rev. 2 requirements accommodates the Relaxed Axial Offset Control (RAOC) Methodology since NUREG-1431 Rev. 2 incorporates the Relaxed Axial Offset Control (RAOC) Methodology in the NUREG format.

In NUREG-1431 Rev. 2, SR 3.2 1 1 is the surveillance requirement that verifies the equilibrium FQ, F_Q^C (Z), requirement. The term F_Q^C (Z) is presented in the Kewaunee TS as the "F_Q^N (Z) equilibrium relationship". That requirement is satisfied by the proposed Kewaunee TS 3.10 b 1.A change

NUREG-1431 Rev. 2, SR 3.2 1 2 is the surveillance requirement that verifies the transient FQ, F_Q^W (Z), requirement It assures that there is at least 2% margin to the K(Z) limit curve when performing the surveillance for the F_Q^W (Z) If two successive maps indicate that there is a decrease in the margin to the limit, either a penalty factor is added to the transient FQ to assure that there will remain margin to the limit up until the next time surveillance is performed or more frequent maps are performed to assure that margin to the limit is maintained. That requirement is satisfied by the proposed TS 3.10.b.6 C. Instead of using the equation of the NUREG, wording is provided that is consistent with terminology currently used by Kewaunee to reflect this requirement. The F_Q^W (Z) term of the NUREG and the Kewaunee phrase "F_Q^{EQ} (Z) transient relationship" have the same meaning.

For the FQ action requirements, additional proposed changes are recommended The NUREG-1431 Rev 2 Action A is presented as proposed change TS 3.10 b 3 which replaces the previous recommended TS 3.10 b 3. The Kewaunee proposed action statement provides the same requirements as the NUREG. Instead of providing it in tabular form, it is provided in paragraph form consistent with the current Kewaunee technical specification format. NUREG-1431 Rev 2 Action B is presented as proposed change TS 3 10 b 7 which replaces the previous recommended TS 3.10 b 7. Its presentation follows the same logic as that of Action A Since the paragraph form is used, NUREG-1431 Rev 2 Action C is added to Kewaunee TS 3.10.b.3 and TS 3 10.b 7 since the same action is required if the equilibrium or the transient surveillance requirement is not met

g. Proposed TS 3 10 b 6 C provides the actions addressing an increase of F_Q^N by 2% or more, when compared to the last power distribution map Item 2 6 3 of the Draft KNPP core operating limits report (COLR) states, "The penalty factor for TS 3 10.b 5 C.i (which appears to be a typo) shall be 2% " Please discuss the magnitude and method of the penalty applied and actions if the increase is greater than 2%

NMC Response: The Core Operating Limits Report (COLR) Item 2 6 3 does contain a typo and will be corrected to read "The penalty factor for TS 3.10.b.6.C i shall be 2%".

The peak pin power relationship as described in TS 3 10.b.6 C is being removed TS 3.10 b 6.C is being revised to be consistent with NUREG-1431 REV 2 and the Relaxed Axial Offset Control (RAOC) Methodology as show in the response to item f.

h. Proposed TS 3 10.b 8 provides the requirements regarding Axial Flux Difference (AFD) Limits Please provide the technical bases for allowing 15 minutes to restore AFD to within limits prior to reducing power to less than 50% RTP and for TS 3 10.8 B, which provides the actions and completion times for the AFD alarms being inoperable These actions are not included in STS or KNPP current TS.

NMC Response· NMC will remove the 15-minute allowance to restore AFD and incorporate that portion of TSTF 110, Revision 2 that allow for the removal of the AFD alarm requirements and adds a surveillance requirement to verify AFD within limits weekly. (see Attachment C)

i KNPP has adopted much of the STS Section 3 2 Power Distribution Limits. Please discuss why KNPP is not revising the current TS for Quadrant Power Tilt in accordance with the STS.

NMC Response To change the QPTR TS was not necessary for the transition to Westinghouse 422V+ fuel or for RAOC, therefore NMC did not request this change.

j Proposed TS 3.10 m changes the Reactor Coolant flow from a loop flow of ≥93,000 gpm to a total flow rate of ≥178,000 gpm Please provide the technical basis for this change

NMC Response The change is made consistent with WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report" Based on that methodology and in response to RAI Question 2, the minimum limit for RCS total flow based on a staff approved analysis is retained in the Technical Specification The cycle specific RCS total flow rate is relocated to the COLR Note that the analysis method for Westinghouse is based on total flow and not individual loop flow

Attachment 3 Questions
Westinghouse Report - Technical Basis for Transitioning to V422+ Fuel

1. The licensee's submittal references the NRC staff's Safety Evaluation Report (SER) regarding the acceptability of VANTAGE+ fuel (WCAP-12610-P-A) With respect to oxidation and crud buildup, the staff's conclusions were based on a limited amount of available ZIRLO oxidation data As such, page 9 of the SER includes the requirement that, "Future plant applications of VANTAGE+ design must also demonstrate that the ZIRLO oxidation data is applicable for these applications " Please discuss the review performed to ensure this data is acceptable for application at KNPP.

NMC Response As presented to the NRC staff during semi-annual fuel performance update meetings between Westinghouse and the NRC, ZIRLO™ corrosion is being tracked against the licensed corrosion model currently used for ZIRLO™ application In performing bounding transition core analysis, the calculated corrosion for the ZIRLO™ clad fuel for KNPP meets the Westinghouse internal limit of < 100 microns on a best estimate nominal basis. KNPP application matches many other plants in the Westinghouse corrosion database, thus, there is no reason to consider any different results With respect to crud, crud deposition or existence is not related to the clad material Crud is an aspect of the primary chemistry and the "steaming rate" of the fuel. Since the fuel is expected to have improved thermal mixing and a reduced "steaming rate", it is not expected that crud will be an issue, especially since KNPP has not seen problems with crud in previous cycles

2. Section 1 6 - The licensee states that the reactor trip setpoints for the new analyses performed assuming the uprated power will also be bounding because they were generated for the uprated power but will be used for the current NSSS Power level. The licensee concludes that they are therefore more restrictive than the current trip setpoints that are being used This statement implies that reactor trip setpoints are being revised, which requires a Technical Specification change. Which setpoints are being changed as part of this license amendment request? Also, please provide the technical basis for the conclusion that all reactor trip setpoints used in the new uprate power analyses are more restrictive than the current setpoints.

NMC Response: The only technical specification reactor trip setpoint values that are being changed as part of this license amendment request (LAR) are the Overtemperature ΔT and the Overpower ΔT setpoints The specific revisions to these two reactor trip setpoints are shown in the table below. Since these two setpoints have been relocated to the COLR (Reference 1) a technical specification change is not required The revised Overtemperature and Overpower ΔT setpoints and the basis for the changes are documented in the safety analysis.

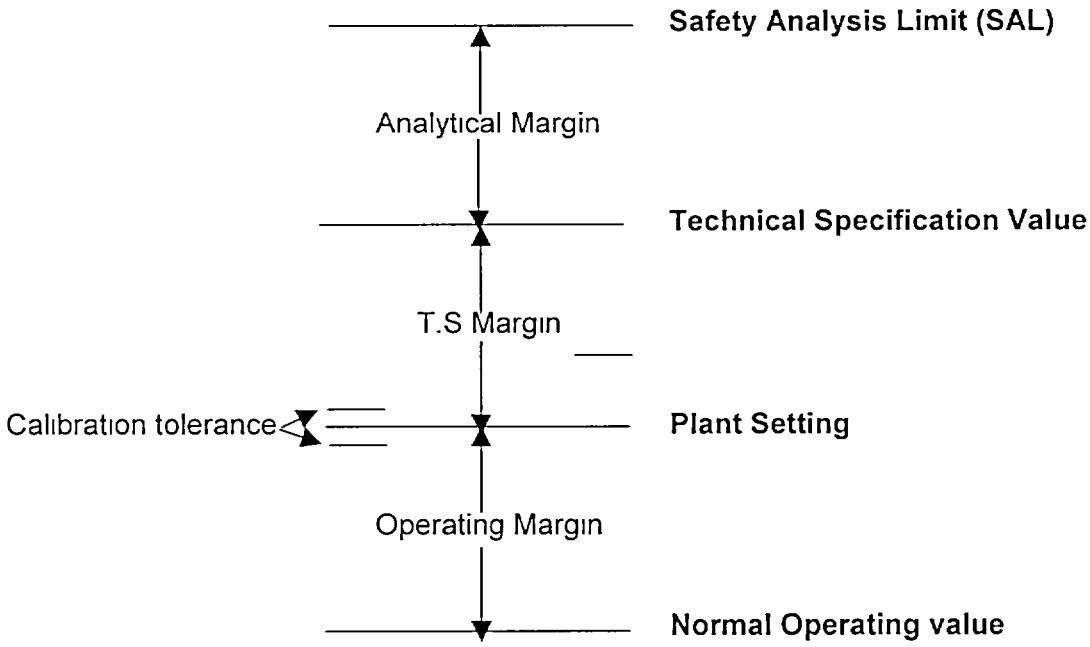
Parameter	Plant Setting		COLR Value		Safety Analysis	
RPS Setpoints	Cycle 25	Cycle 26	Cycle 25	Cycle 26	Cycle 25	Cycle 26
OTDT						
Bias K1	1.0975	**1.1875**	1 11	**1.20**	1.19	**1.30**
Gain on T_{avg} mismatch K2	0.0090	**0.015**	0 0090	**0.015**	0.0090	**0.015**
Gain on	0.000566	**0.00072**	0.000566	**0.00072**	0.000566	**0.00072**

Parameter	Plant Setting		COLR Value		Safety Analysis	
RPS Setpoints	**Cycle 25**	**Cycle 26**	**Cycle 25**	**Cycle 26**	**Cycle 25**	**Cycle 26**
pressure K3						
Lead time constant on T_{avg} mismatch T_1	30 sec	30 sec	30 sec	30 sec	30 sec	30 sec
Lag time constant on T_{avg} mismatch T_2	4 sec	4 sec	4 sec	4 sec	4 sec	4 sec
T'	567 3°F	**572°F**	567.3°F	**≤ 573.0°F**	568 8°F	**≤ 573.0°F**
P'	2235 psig	2235 psig	2235 psig	2235 psig	2235 psig	2235 psig
FΔI	qt-qb – 12%, +9% f(dI)=0%/% qt-qb > +9% f(dI)=2.5 %/% qt-qb < -12% f(DI) = 1.5%/%	**qt-qb – 20.5%+10.5% f(ΔI)= 0%/% qt-qb > +10.5% f(ΔI)= 0.96%/% qt-qb < 20.5% f(ΔI) = 0.86%/%**	qt-qb – 12%, +9% f(dI)=0%/% qt-qb > +9% f(dI)=2.5 %/% qt-qb < -12% f(DI) = 1.5%/%	**qt-qb – 22%,+12% f(dI)= 0%/% qt-qb > +12% f(dI)= .96%/% qt-qb < -22% f(dI) = .86%/%**	Qt-qb – 12%, +9% f(dI)=0%/% qt-qb > +9% f(dI)=2.5 %/% qt-qb < -12% f(DI) = 1.5%/%	**qt-qb – 22%,+12% f(dI)= 0%/% qt-qb > +12% f(dI)= .96%/% qt-qb < -22% f(dI) = .86%/%**
OPDT						
Bias K4	1.0875	**1.0825**	1.10	**1.095**	1.19	**1.16**
Gain on T_{avg} K5	0 0275	0.0275	0.0275	0 0275	0 0275	0 0275
Gain on T_{avg} mismatch K6	0 002	**0.00103**	0.002	**0.00103**	0 0002	**0.00103**
Rate lag time constant on T_{avg} T3	10 sec	10 sec	10 sec	10 sec	10 sec	10 sec
FΔI	Qt-qb – 12%, +9% f(dI)=0%/%	**0%/% for all ΔI**	qt-qb – 12%, +9% f(dI)=0%/%	**0%/% for all ΔI**	qt-qb – 12%, +9% f(dI)=0%/%	**0%/% for all ΔI**

Parameter	Plant Setting		COLR Value		Safety Analysis	
RPS Setpoints	Cycle 25	Cycle 26	Cycle 25	Cycle 26	Cycle 25	Cycle 26
	qt-qb > +9% f(dI)=2 5 %/% qt-qb < - 12% f(DI) = 1.5%/%		qt-qb > +9% f(dI)=2.5 %/% qt-qb < - 12% f(DI) = 1 5%/%		qt-qb > +9% f(dI)=2 5 %/% qt-qb < - 12% f(DI) = 1.5%/%	

The conclusion that all reactor trip setpoints used in the new uprate power analyses are more restrictive than the current setpoints is meant to convey the following Since all reactor trip plant setpoints (except the ΔT trips) retain the existing setpoint values and the normal operating values (e g , power, T_{avg}, T_{out}, ΔT) have increased, the operating margin present in the uprated plant has been effectively decreased. The decrease in operating margin implies that the setpoints are more restrictive for the uprated plant than for the non-uprated plant. Refer to the figure below for clarification

The ΔT trip parameters have been revised as stated above, in some instances increasing and in some instances decreasing the operating, technical specification, and analytical margins It cannot be simply stated that the revised ΔT setpoints represent more or less operating margining, only that the technical specification, analytical and safety margins are adequate, as demonstrated by the safety analyses



It should be noted that the reactor protection and engineered safety features setpoints modeled in the uprated safety analyses are also adequate to provide protection in the non-uprated plant It can be stated with confidence that if repeated at a lower rated power level, the safety analyses would result in the same or additional margin to the steady-state and transient acceptance criteria Therefore, all margins are adequate and the setpoints provide adequate protection to all applicable limits (safety, analytical and technical specification) for both the uprated and non-uprated plant

References:

1. NMC Letter to NRC, from Mark E. Warner to Document Control Desk, entitled "License Amendment Request 185 To The Kewaunee Nuclear Power Plant Technical Specifications, "Core Operating Limits Report Implementation", Letter NRC-02-064, dated July 26, 2002 (TAC No MB5717)

3. To calculate the uncertainties for the RTDP methodology, the analyses in WCAP-15591 (the KNPP specific RTDP analysis) assumed an uprated power of 1757 MWt (NSSS) and 54F Replacement Steam Generators. The licensees submittal proposes an uprated power level of 1780 MWt (NSSS) Please quantify any differences in the WCAP-15591 RTDP analysis results for an uprate power of 1780 MWt. What impact does this have on the DNBR limits?

NMC Response. WCAP-15591, Rev 0 is for an uprated power of 1757 MWt (NSSS). WCAP-15591, Rev.1 addresses an uprated power of 1780 MWt (NSSS) The uncertainties of WCAP-15591, Rev.0 are conservative for an uprated power of 1780 MWt (NSSS)

4. Table 1-2 in the Westinghouse report specifies an RCS design flow of 89,000 gpm/loop The licensee has proposed a new updated final safety analysis report (UFSAR) Table 14 0 3 which provides a table of nominal values for non-LOCA accident analyses for RTDP methodology, and applies uncertainties to these nominal values for STDP methodology However, the way that the RCS flow value from Table 1-2 is used is not consistent with the way that the Tavg and RCS pressure values are used. Please discuss the technical basis for using 93,000 gpm value in this table. Why is uncertainty not applied to the 89,000 gpm nominal value for the STDP methodology? When comparing Table 1-2 of the Westinghouse report with new USFAR Table 14.0.3, it appears that a non-conservative RCS flow value may have been used in the safety analyses. Please discuss this discrepancy and how it relates to the proposed minimum RCS flow TS value

NMC Response: The reactor coolant minimum measured flow (MMF) is 186,000 gpm total (93,000 gpm/loop) as indicated in the COLR item 2.11.3 The reactor coolant thermal design flow (TDF) is 178,000 total (89,000 gpm/loop) as indicated in Table 1-2 of the Westinghouse Report and TS 3.10.m. The MMF is used in Revised Thermal Design Procedure (RTDP) analyses and can be regarded as the RTDP nominal value of the reactor coolant flow. The TDF is used in Standard Thermal Design Procedure (STDP) analyses and is equal to the MMF minus the RTDP flow uncertainty of 4.3% as indicated in Table 1-1 of the Westinghouse Report (KNPP LAR 187 submittal, Attachment 4)

In the RTDP methodology, the RTDP nominal flow (i.e., MMF) is used and the flow uncertainty is statistically combined with the pressure, temperature, and power uncertainties to define the DNBR safety analysis limit to which the resultant DNBR is compared In the STDP methodology, the MMF minus the RTDP flow uncertainty is used (i.e., TDF) and the resultant DNBR is compared to the DNBR safety analysis limit

Compliance with the proposed COLR minimum measured flow value of 186,000 gpm total ensures that both the thermal design flow (TDF) and the minimum measured flow (MMF) are bounding and conservative for the analyses in which they are used. Note that since the MMF is well below the plant nominal reactor coolant flow of 99,000 gpm/loop there is significantly more conservatism in the flow assumption than for other parameters (e g , power and pressure) for which plant nominal values are used for the RTDP nominal values

5. The NRC staff issued the SER for WCAP-11397-P-A, "Revised Thermal Design Procedure," in January 1989 Please provide a discussion which demonstrates how each of the seven restrictions outlined in the Staff's SER for this methodology are satisfied considering the proposed new fuel to be used at KNPP.

NMC Response: The seven conditions were considered in the safety evaluation. Each of the seven conditions is addressed below, and a reference is provided for the specific section in the safety evaluation or other appropriate documentation where the condition is discussed

Condition 1
Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

Response:
Sensitivity factors were evaluated using the WRB-1 DNB correlation and the VIPRE code for parameter values applicable to the Kewaunee 422V+ fuel. These sensitivity factors were used to determine the maximum Design Limit DNBR for Kewaunee The resultant Design Limit DNBR is included in the Kewaunee LAR 187 submittal.

Condition 2:
Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397-P-A outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

Response
See Response to Condition 1 above, as well as Section 4 2 of the Kewaunee LAR 187 submittal for the justification of the use of the WRB-1 correlation and the VIPRE-01 code for Kewaunee application. Additional discussion on the use of WRB-1 can be found in the response to RAI Question 21

Condition 3:
If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

Response:
Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) have been shown to be valid for the combination of WRB-1 and the VIPRE code which was used for the application of RTDP to the Kewaunee 422V+ fuel The sensitivity factors, operating parameters, and the VIPRE model used in this application do not differ significantly from those used in WCAP-11397-P-A

Condition 4:
Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained

Response:
The plant specific variances and distributions for this application were justified in the proprietary report WCAP-15591, Rev 1.

Condition 5:
Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions

Response
Nominal initial conditions were only applied to DNBR analyses, which used RTDP

Condition 6:
Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation

Response:
Bounding nominal conditions were used in the DNBR analyses that were based on RTDP

Condition 7
The code uncertainties specified in Table 3-1 (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP

Response:
The code uncertainties specified in Table 3-1 of WCAP-11397-P-A were included in the DNBR analyses using RTDP.

6 The proposed new V422+ fuel design for KNPP incorporates VANTAGE + features, PERFORMANCE + features and other KNPP specific features (0 422 inch OD fuel rod and instrumentation tube and new OFA style mid-grid). The NRC staff has reviewed the VANTAGE + features in WCAP-12610-P-A The PERFORMANCE + features were evaluated by Westinghouse in SECL-92-305 under 10CFR50.59 guidelines. Please explain why the PERFORMANCE + features were not evaluated under the NRC approved Fuel Criteria Evaluation Process?

NMC Response: The PERFORMANCE + features were developed and reviewed under 10 CFR 50.59 guidelines since FCEP (WCAP-12488-A) was not licensed by the NRC at the time PERFORMANCE + features were developed.

7. The Westinghouse report states that the 422V+ and the 14x14 Framatome/ANP designs are mechanically and hydraulically compatible with each other. Westinghouse bases this conclusion on a referenced report (PD2-01-46, Rev. 1, "Kewaunee Fuel Transition Work Report, Revision 1 to Fuel Assembly Compatibility Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assembly," November 1, 2001) Please list the technical considerations (items) that were evaluated in this referenced report and discuss the methodologies applied Please include any references to NRC-approved methodologies.

NMC Response: Regarding hydraulic compatibility, Section 4.3 of Attachment 4 to LAR 187 discusses what was evaluated to determine hydraulic compatibility. The determination of hydraulic compatibility is based on experienced based parameter comparisons. The component loss coefficients, rod bundle geometry, and grid axial elevations have been compared between the Westinghouse 422V+ design and the Framatome/ANP design This comparison shows that the designs are compatible as supported by the fuel assembly crossflow results (see response to RAI Question 23 Part a below) Analysis has demonstrated that no significant crossflow induced vibration will result in the transition core A second area of hydraulic compatibility that was assessed regards fuel assembly lift forces This area is discussed in the response to RAI Question 23 Part b below

The mechanical compatibility evaluation included in the referenced compatibility report (PD2-01-46, Revision 1) consists of a geometric evaluation of grid overlap (compatible grid axial location) and a geometric evaluation of anti-snag features (compatible fuel handling) Acceptability of the Westinghouse fuel and loads imparted on co-resident fuel during seismic LOCA events has been documented in the safety analyses. The codes used for seismic LOCA analysis and their pedigree are discussed in the response to a RAI Question 14 below

8. Regarding compatibility of the 422V+ and Framatome/ANP fuels with respect to crossflow, Westinghouse provides a criterion that there should be outer grid strap overlap between any two fuel assemblies in the core throughout their life in the core Please provide further clarification regarding this statement as follows

   a What is meant by outer grid strap overlap and the quantity of overlap required?

NMC Response. The outer grid strap of each mid-grid shall maintain an overlap between mid-grids of adjacent fuel assemblies even if one assembly is a fresh feed assembly compared to a twice or thrice burned assembly The overlap accounts for thermal and irradiation effects of assembly growth/expansion The overlap is based on maintaining a minimum of 20% overlap between mid-grid inner-straps Since the outer-straps are slightly larger than the inner-straps, appropriate overlap with the outer-straps will always be maintained This is an internal Westinghouse design criterion used to ensure crossflow and rod vibration are minimized

b The technical basis for this criteria?

NMC Response The technical basis for this criterion is to minimize/prevent hydraulic crossflow between fuel assemblies which could lead to added DNB penalties and would also lead to additional concerns associated with rod vibration The criterion also ensures appropriate load sharing between mid-grids with respect to seismic/LOCA loads

c. A reference to the NRC-approved methodology applied to determine that this criteria is satisfied for a core consisting of 422V+ and Framatome/ANP fuel

NMC Response: The criterion is an internal Westinghouse design criterion It has been used on all fuel designs including those reviewed by the NRC in WCAP-9500-A, WCAP-10444-P-A and WCAP-12610-P-A (OFA, VANTAGE 5 and VANTAGE + fuel products) The criterion is independent of Westinghouse fuel type or array geometry.

9. Section 2 2 3 of the Westinghouse report states that, "The diameter of the 422V+ guide thimbles is 0.015 inches smaller than that of the 14x14 Framatome/ANP design The 422V+ guide thimble inner diameter provides a minimum diametral clearance of 0.0088 inches (under worst case conditions) for control rods supplied by Westinghouse " What are the accepted design criteria established for minimum clearance, and please define the worst-case conditions Is rod cluster control assembly (RCCA) insertion time impacted by this change?

NMC Response: The design criterion of concern is rod drop time To ensure appropriate rod drop times, certain minimum clearances must be maintained when accounting for the hydraulic resistance encountered by the RCCA during insertion. The worst scenario is as defined in the report

As reported in the KNPP LAR 187 submittal, a DROP analysis was performed to determine the RCCA drop time for the 422V+ fuel upgrade and uprating to 1772 MWt The maximum RCCA drop time with seismic allowance was calculated to be 1 59 seconds, which satisfies the current Technical Specification limit of 1 8 seconds Therefore, RCCA insertion times are acceptable.

10. Section 2.4.1 of the Westinghouse report addresses fuel rod design criteria. The report addresses a list of various criteria, but does not address all criteria listed in NUREG-0800, Standard Review Plan (SRP) Section 4.2. Please provide the basis for choosing only the criteria evaluated in Section 2.4 1 Provide additional evaluations as necessary

NMC Response: Westinghouse fuel rod design criterion exceeds those specified in the SRP. (See Below)

Standard Review Plan

1. Stress, Strain
2. Clad Fatigue
3 Oxidation, hydrating, cladding overheating
4. Rod Growth
5 Rod internal Pressure
6. Cladding Collapse
7. Fuel centerline melt

Westinghouse Fuel Rod Design Criteria

1. Clad stress, transient strain and steady state strain
2. Clad Fatigue
3. Oxidation, hydrating, cladding overheating
4. Rod growth
5 Rod internal pressure (Gap reopening and DNB propagation)
6 Cladding Collapse
7 Fuel centerline melt
8 Clad free standing
9 End plug weld integrity
10. Plenum clad support

11. In Section 2.4.1 of the Westinghouse report regarding rod internal pressure, it is stated that, "The design limit for Condition II events is that DNB propagation is not extensive, that is, the process is shown to be self-limiting and the number of additional rods in DNB due to propagation is relatively small." NUREG-0800, SRP Section 4.2 specifies the acceptance criteria for evaluation of fuel design limits. One of the criteria provides assurance that there be at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience DNB during normal operation or anticipated operational occurrences (Condition II events) Does this contradiction impact the conclusions reached in the Westinghouse report?

NMC Response: No Westinghouse ensures that DNB does not occur on a 95/95 basis for both Condition I and II events based on the critical heat flux correlation limit (e g , WRB-1) The criterion specified above for the fuel rod design is based on the premise that DNB has occurred within a fuel assembly. With fuel rods operating over system pressure, the criterion looks at whether rods above system pressure and adjacent to a rod in DNB would result in any appreciable number of fuel rods entering into DNB through propagation effects.

12. In Section 2.4.1 of the Westinghouse report regarding clad oxidation, please provide the expected clad oxidation values for the 422V+ fuel design operation at KNPP.

NMC Response The limiting clad oxidation results for the uprated core power level of 1772 MWth are summarized below

CRITERION	LIMIT	KNPP RESULTS
CLAD OXIDE THICKNESS	≤ 4.00 MILS	2.0 MILS
99% UB CLAD OXIDE THICKNESS	≤ 6.00 MILS	3.8 MILS
CLAD STEADY STATE TEMPERATURE	≤ 780 DEG F	685 1 DEG F
CLAD TRANSIENT TEMPERATURE	≤ 850 DEG F	705 7 DEG F
CLAD HYDROGEN PICKUP	≤ 600 PPM	319.9 PPM

13 In Section 2.5 of the Westinghouse report regarding Seismic/LOCA impact on fuel assemblies, the analysis results demonstrated adequate grid load margin for all fuel assemblies except the fuel assemblies on the periphery of the 13 assembly row in the limiting mixed Condition I Please discuss the various core configurations evaluated as part of this analysis and identify which configuration produces the limiting condition.

NMC Response Two limiting mixed core configurations were evaluated for Kewaunee Seismic/LOCA analysis

- Mixed Condition I is W, F, F, F, ..., F, W
- Mixed Condition II is F, W, W, W, ., W, F
  Where W = Westinghouse fuel
  F = Framatome fuel

The grid impact load results show that the Mixed Condition I configuration is the limiting condition The analysis results demonstrated adequate grid load margin for all fuel assemblies except the fuel assemblies on the periphery of the 13 assembly row in the limiting mixed Condition I.

The use of Leak Before Break methodology was utilized in the input to the LOCA Hydraulic Forces Analysis. The LOCA Hydraulic Forces Analysis is used, in part, as input to the Vessel Internals Dynamic Analysis and in turn the information is used in the Fuel Structural LOCA and Seismic analysis This analysis result is then used in determination of adequate grid load margin.

Further, the grid impact loads of the 422V+ homogeneous core evaluated for the LOCA and Seismic lateral loading and combined by the SRSS method identified in SRP 4 2 are less than the allowable limit Thus, the core coolable geometry is maintained.

The vessel and internals are qualified on the basis of branch line breaks, notably the accumulator line and pressurizer surge line, as allowed under leak-before-break (LBB), (references 1 through 5) Although KNPP has previously been approved to eliminate the pressurizer surge line under LBB (references 6 and 7), the pressurizer surge line break was chosen for the LOCA hydraulic forces analyses to conservatively represent all possible hot-leg branch-line breaks.

References:

1. WCAP-11411, Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee, Rev. 1 (Proprietary), and WCAP-11410, Rev 1 (Non-Proprietary), April 1987
2 WCAP-11619, Additional Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Kewaunee, (Proprietary), and WCAP-11620 (Non-Proprietary), September 1987
3 K E. Perkins (NRC) to D. C. Hintz (WPS), K-88-32, Application of Leak-before-Break Technology as a Basis for Kewaunee Nuclear Power Plant Steam Generator Snubber Reduction, February 16, 1988.
4. J. G. Giitter (NRC) to D C. Hintz (WPS), K-88-50, March 18, 1988
5. WCAP-15311, Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Nuclear Power Plant after SG Replacement (Proprietary), June 2000
6. WCAP-12875, Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Kewaunee Nuclear Plant, June 1991
7. A G. Hansen (NRC) to C. A Schrock (WPS), K-92-005, January 3, 1992

14 In Section 2.5.1 of the Westinghouse report, three computer codes are identified for evaluating the fuel assembly and reactor core models These codes are NKMODE, WECAN and WEGAP Please provide a reference to the NRC approval for these codes. If not NRC-approved, please provide information regarding control of these codes, particularly with respect to quality assurance, benchmarking, validation and verification If used to evaluate Framatome ANP fuel, justify their applicability.

NMC Response WEGAP was approved by the NRC as documented in the Safety Evaluation Report (SER) for WCAP-9401. It is further stated in the SER that the NRC was in the process of reviewing WECAN and that a favorable approval was expected based on the benchmarks conducted by the NRC for WECAN. To date the NRC has never issued Westinghouse the SER for WECAN. However, Westinghouse has conducted benchmark calculations for all three codes. All three codes are governed by Westinghouse software control which requires validation and verification. The codes are all governed by the Westinghouse Quality Management System (QMS) which is Appendix B certified and has been audited and approved by the NRC.

The FEA models for NKMODE and WEGAP were generated in accordance with the Framatome fuel assembly design, assumptions and testing information.

NKMODE was used to generate the following·

- The first five natural frequencies of the Framatome fuel and mode shapes of the Westinghouse fuel assembly
- The tested fuel assembly damping coefficients and span masses used to generate the lumped mass FEA model (five masses)

The WEGAP model for the Framatome fuel assembly was generated to:

- Combine the NKMODE results and the fuel assembly and grids structural information of the Framatone fuel, and the fuel assembly assumptions and grids test information of the Framatone fuel.

15 Section 3 1 of the Westinghouse report states that standard nuclear design analytical models and methods (Westinghouse computer codes and methods) accurately describe the neutronic behavior of the 422V+ fuel design Please discuss the codes and methods applied to evaluate the current Framatome ANP fuel through the transition cores at KNPP Provide technical justification for the applicability of these codes and methods

NMC Response: The neutronic behavior of the Framatome-ANP fuel through the transition cores is accurately described by the standard nuclear design analytical models, codes (the ALPHA/PHOENIX/ANC code system) and methods of Westinghouse. The generic verification and validation basis for the Westinghouse models and methods is extensive and encompasses various Westinghouse fuel designs with different fuel rod diameters, cladding dimensions, assembly designs, core sizes, etc

The nature and magnitude of the differences between the various Westinghouse fuel designs in the qualification basis is significantly greater than the differences between the Westinghouse 422V+ fuel and the Framatome-ANP fuel (see table below) Moreover, the differences between the 422V+ fuel and the Framatome-ANP fuel that impact neutronic behavior are even less significant and are dominated by the typical variations in the uranium loading, enrichments, and number and type of burnable poisons that are modeled Finally, comparisons of the measured and predicted (i.e , predicted with NMC standard methods) nuclear design data from previous KNPP cycles with the Westinghouse predicted data are used to verify that the Westinghouse models and methods accurately describe the Framatome-ANP fuel neutronic behavior.

Characteristic	Westinghouse 422V+	Framatome Heavy HTP
Fuel Rod O D (in)	0.422	0.424
Clad Thickness (in.)	0.0243	0.0250
Pellet Diameter (in.)	0.3659	0.3670
Fuel Rod Pitch (in)	0.556	0.556

The mechanical behavior of the Framatome-ANP fuel is evaluated by Framatome-ANP using the NRC approved models and methods of Framatome-ANP (References 1–3) The evaluation is performed for each cycle containing Framatome-ANP fuel. For Cycle 26 the evaluation has been completed and considers the transition core effects and the power uprate.

The thermal-hydraulic behavior of the Framatome-ANP fuel is evaluated by NMC using the NRC approved models and methods (Reference 4) for KNPP. These models and methods consider the transition core effects and the power uprate. Note that the presence of the 422V+ fuel improves the thermal margins for the Framatome-ANP fuel due to the flow increase experienced by the Framatome-ANP fuel caused by the higher overall loss coefficient in the 422V+ fuel.

References:

1. XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup", Exxon Nuclear Company, October 1986
2. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU", Advanced Nuclear Fuels Corporation, December 1991
3. EMF-92-116(P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs", Siemens Power Corporation, February 1999
1. U S Nuclear Regulatory Commission Letter, entitled "Kewaunee Nuclear Power Plant – Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC No MB0306)", dated September 10, 2001

16. Section 3.1 of the Westinghouse report states that marked-up changes to the Technical Specifications are included in Appendix A Section 3.6 states that the TS changes which impact the nuclear design for 422V+ fuel are modifications to the protection trip setpoints (summarized in Section 5.1). The Westinghouse report does not contain an Appendix A and Section 5.1 does not summarize the protection trip setpoints Please provide this information.

NMC Response: Section 3 1 of Attachment 4 to LAR 187 references Appendix A Appendix A is a part of the RTSR proprietary version sent to NMC by Westinghouse From this report NMC reviewed the TS changes in Appendix A, suggested by Westinghouse, modified the KNPP TS, and then submitted these KNPP TS changes to the NRC as attachments 2 and 3 to KNPP LAR 187 request. Therefore, although Appendix A is not included in the LAR submittal the TS changes necessary for the transition to Westinghouse fuel are included.

17. Section 3.2 of the Westinghouse report states that, "The effects of extended burnup on nuclear design parameters have been previously discussed in WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," and that discussion is valid for the anticipated 422V+ design discharge burnup " What is the technical basis for this conclusion? The 422V+ product design is extended to a lead rod burnup of 75,000 MWD/MTU. WCAP-10125-P-A discusses a lead rod average burnup of 60,000 MWD/MTU.

NMC Response The fuel rod lead rod burnup limit specified in WCAP-10125-P-A is based on the mechanical aspects of the fuel rod/assembly design and material (i e , irradiation growth and corrosion/hydriding). The nuclear design aspects of the fuel are physics-based, first principal criteria and are not specifically tied to burnup. Isotopic depletion, which is physics based can be taken to much higher burnups than are currently licensed. A majority of the nuclear design parameters are beginning-of-life (BOL) limiting and thus are not impacted by high burnup. Since the nuclear aspects of the design are based on the fuel pellets, with minor consideration of the cladding (i.e , effects of niobium), the nuclear aspects of the fuel are independent of the fuel assembly design (other than geometric considerations such as rod pitch, rod diameter, etc). The current licensed lead rod average burnup design limit for Westinghouse fuel is 60 GWD/MTU with extension of that burnup limit to 62 GWD/MTU based on Appendix R of WCAP-12488-A. Generating nuclear parameters to 75 GWD/MTU can be done and the fuel assembly can be designed to a 75 GWD/MTU burnup based on mechanical aspects of the fuel such as irradiation growth and sizing of the fuel assembly/fuel rod However, the 422V+ design will be limited to the current licensed burnup limit until such time as 75 GWD/MTU design criteria and limits have been established and licensed with the NRC.

18. Section 3.4 (Attachment 4 of KNPP LAR 187) of the Westinghouse report states that four core models were developed and used in the nuclear design analyses performed. However, this section only discussed three of the models, first transition cycle, second transition cycle and a third "all 422V+" core Please discuss the fourth core design evaluated Also, were NRC-approved methods employed in determining the transition core configurations evaluated or is this based on standard reload methodology?

NMC Response: Only three core models were developed The last sentence of the 2nd paragraph of Section 3 4 (Attachment 4 of KNPP LAR 187) should state, "Three core models were developed and used for the majority of calculations performed here". Only approved NRC codes in compliance with all applicable limitations and restrictions were used in the evaluation of the transition cores. (See Attachment 3 question 35 of this attachment)

19. Section 3 5 of the Westinghouse report addresses power distributions and peaking factors The terminology used here is not clear. Westinghouse refers to the factor $F_Q^N(z)$ as the total peaking factor in this section and in Figure 3 8 Typically, the total peaking factor is identified as F_Q^N, and is not a function of height (z) Is Figure 3 8 shown for a particular axial location? Please clarify the terminology used and verify that the total peaking factor limit will be met with the new 422V+ fuel design

NMC Response. Figure 3 8 is a plot of the best estimate total peaking factor in the core as a function of cycle burnup These values were taken directly from the 3D ANC code and plotted without uncertainty. These values are at various axial and radial locations throughout the cycle. The axial designation "(z)" should be deleted from section 3 5 and Figure 3 8

The total peaking factors for Westinghouse 422V+ and FRA ANP fuel, after accounting for uncertainty and the effects of numerous xenon shapes from various core power maneuvers, will meet the limit of 2 50 and 2 35, respectively.

20. With respect to Nuclear Design, please discuss how the acceptance criteria specified in NUREG-0800, SRP Section 4 3 are ensured to be satisfied for the 422V+ fuel design

NMC Response: The Nuclear Design acceptance criteria of NUREG-0800, SRP Section 4.3 are satisfied since:

a) NRC approved methods are used (See Attachment 3 Question 35 of this attachment)
b) Approved codes are employed (See Attachment 3 Question 35 of this attachment)
c) All applicable Westinghouse design limits and acceptance criteria are satisfied.
d) All applicable Tech Spec limits will be satisfied during surveillance for the 422V+ fuel

21. Section 4 2 of the Westinghouse report states that the use of the WRB-1 DNBR correlation with a 95/95 correlation limit of 1 17 is applicable for the 14x14 V422+ fuel assemblies The NRC staff SER for WCAP-8762. "New Westinghouse Correlation WRB-1 For Predicting Critical Heat Flux in Rod Bundles With Mixing Vane Grids," states that the WRB-1 correlation may be used for 15x15 and 17x17 optimized fuel assembly design. The justification provided by Westinghouse for the application of WRB-1 and its associated limit to 14x14 fuel assemblies is based on Westinghouse responses to requests for additional information (RAI's) on WCAP-8691, "Fuel Rod Bow Evaluation " RAI responses do not constitute NRC staff approval. Additionally, it is not apparent where this issue was discussed in the referenced RAI responses. Please provide the technical basis for application of the WRB-1 correlation and correlation limit of 1 17 for the 14x14 V422+ fuel assembly designs In your response, please discuss grid design differences between the fuel types.

NMC Response. The NRC staff approved WRB-1 for both the 14x14 OFA and the 15x15 OFA with a correlation design limit of 1 17. Refer to WCAP8762, Supplement 1, page 3 and 4 of SER dated June 1984.

> "The 14x14 OFA CHF data can be incorporated into the total R-grid population The staff concludes that WRB-1 is applicable to the 14x14 OFA with a DNBR limit of 1 17."
>
> "We find that the WRB-1 correlation is acceptable for application to both 14x14 OFA and 15x15 OFA with a minimum DNBR limit of 1.17 "

With respect to the grid design for the 422V+ fuel product, it has been licensed via the Fuel Criteria Evaluation Process (FCEP). Refer to NPL 97-0538 (WEPCO)/ CAW-97-1166 (Westinghouse), Slide Presentation, "14x14, 0.422" OD VANTAGE + (422V+) Fuel Design, Application for Point Beach Units 1 & 2," dated September 9, 1997

22. The licensee is proposing to revise the nuclear enthalpy rise hot channel factor equation by changing the reduction term from [1+0 2(1-P)] to [1+0 3(1-P)] Please provide the technical basis for this change.

NMC Response: The use of 0 3 part power multiplier is consistent with the Westinghouse design methodology and the conservative safety analyses

23 Section 4.2 of the Westinghouse report discusses the hydraulic compatibility of the various fuel assemblies

a Regarding the conclusion that the fuel assembly crossflow is well within the bounding Westinghouse experience base of transition core analysis, please provide quantitative results of the significant parameters of interest which would justify this conclusion For example, the response to this question may include a table comparing ΔP values for the Framatome ANP fuel, 422V+ fuel and Westinghouse OFA fuel

b Westinghouse states that the Framatome ANP fuel assemblies may experience an increase in assembly lift forces of approximately 10% once the Westinghouse 422V+ fuel assemblies are loaded in the core. Please discuss the consequences of this increase and justify why such an increase is acceptable under normal and transient operating conditions.

NMC Response

a. The crossflow velocity profile for a transition core is driven by the local pressure drop differences between the adjacent assemblies of interest These pressure drop differences occur due to differences in the fuel assembly component loss coefficients and rod bundle flow areas (i.e., from geometric differences like fuel rod and thimble tube OD's) Differences in geometric parameters are presented in Table 2-1 of the Attachment 4 to LAR 187 Differences in the fuel assembly loss coefficients are presented in Table 6.4-2 of Reference 1

The significant point of interest for the fuel assembly crossflow is shown by a comparison of the peak crossflow values Comparing Figures 6.4.1 and 6 4 2 of Reference 1 shows that a prior Westinghouse transition core (14 Standard / 14 OFA) has peak crossflow velocity of approximately 2.5 ft/sec while the transition core of interest (Westinghouse 422V+ / Framatome ANP) has a peak crossflow velocity of approximately 1 5 ft/sec. Additionally, a prior Westinghouse transition has higher crossflow over the entire axial domain when compared to the Westinghouse 422V+ / Framatome ANP transition

This comparison of crossflow velocity profiles shows that the fuel assembly crossflow is bounded by Westinghouse experience for 14x14 fuel. For Westinghouse 15x15 and 17x17 fuel, IFM grids have been introduced (assemblies with IFM grids in mid-span locations against assemblies without grids in mid-span locations) which increase the local crossflow velocity to approximately 4.0 ft/sec for transition cores Therefore, the transition of interest for the Westinghouse 422V+ fuel is well within the bounding Westinghouse experience for all fuel designs.

b. The consequence of the increase in lift forces on the Framatome ANP (FANP) assemblies is that the margin to the liftoff force limit is decreased for the FANP assemblies (and increased for the Westinghouse assemblies). The decrease in liftoff force margin for the FANP assemblies under normal and transient operating conditions is acceptable for Cycle 26 based on the results of detailed liftoff force calculations performed by FANP. The Westinghouse assessment is that the limiting case of higher lift force in the Framatome ANP fuel assemblies is acceptable. A more detailed discussion follows

Design and manufacturing differences between the Westinghouse 422V+ assemblies and the Framatome ANP (FANP) HTP Heavy assemblies result in an overall loss coefficient for the Westinghouse design that is approximately 10% greater than the FANP design. Therefore, under similar conditions a Westinghouse assembly will experience a pressure drop through the assembly that is approximately 10% greater than an FANP assembly (Reference 1). A 10% difference in design pressure drop equates to a maximum co-resident flow increase of 4.88% in an FANP assembly (a single FANP assembly in a sea of Westinghouse assemblies), which equates to a 10% increase in liftoff forces ($F_{drag} \sim v^2$).

The consequence of the increase in flow in the FANP assemblies is that the margin to the liftoff force limit is decreased for the FANP assemblies (and increased in the Westinghouse assemblies) In addition, the margin to the DNBR limit is increased for the FANP assemblies (and decreased in the Westinghouse assemblies)

The decrease in liftoff force margin under normal and transient operating conditions for the FANP fuel is acceptable for Cycle 26 based on the results of detailed liftoff force calculations performed by FANP. The decrease in DNBR margin under normal and transient operating conditions for the Westinghouse fuel is acceptable based on the results of the detailed DNB analysis performed in the safety analysis and verified in the reload safety evaluation It should be noted that the decrease in margin for liftoff force in FANP assemblies is remedied by the completion of the transition to Westinghouse fuel, and that the decrease in margin for DNBR in the Westinghouse assemblies is regained when the transition to Westinghouse assemblies is complete

References:

1. Westinghouse Report, entitled "Kewaunee Fuel Transition Work Report, Revision 1 to Fuel Assembly Compatibility Report for the Supply of 14 x 14 Westinghouse 422V+ Fuel Assemblies", PD2-01-46, Revision 1, dated November 1, 2001

24. Section 4.6 of the Westinghouse report discusses the transition core effect and the calculation of a transition core DNBR penalty. The calculated penalty is shown in Figure 4-5 and in the equation in Section 4 6. Figure 4-5 shows the relationship between fraction of 422V+ in the core and the DNBR penalty to be linear, and shows results for two transition core configurations of approximately 10 percent and 55 percent 422V+ fuel loaded in the core. Is this linear relationship based on actual analysis or is it an assumption? If this is based on actual data, then please provide a plot including the data points. If this is an assumption, then please justify that the DNBR penalties to be used for the 1st and 2nd transition cycles are conservative.

NMC Response. At current operating conditions, the linear relationship shown on Figure 4-5 was assumed, this is based on the fact that enough DNBR margin is available (more than 10%) to offset the worst DNBR penalty (4 22%, which conservatively assumes one 422V+ fuel assembly is in the core) for both transition cycles at the current power level. Additional calculations were performed in conformity with the transition core penalty NRC approved methodology (WCAP-11837-P-A) to support the uprated operating conditions Several 3x3 and 5x5 arrays were analyzed and additional penalty was taken to account for variance in the curve fitting The results will be presented in the Kewaunee Power Stretch Uprating Submittal

25. Table 4-4 of the Westinghouse report details the DNBR margins available.

    a. Please show how the DNBR design limit of 1.24 was calculated

    b. The DNBR safety limit of 1.34 was selected to conservatively bound the effects of rod bow, transition core and any other DNBR penalties that may occur, and to provide operating flexibility The margin between the DNBR design and safety limits is shown as 7.46 percent. However, the DNBR penalties listed in this table sum to approximately 8 4 percent Therefore, the retained DNBR margin does not appear to be sufficient Does the DNBR safety limit need to be further increased to account for these effects?

NMC Response:

a The DNBR Design Limit of 1.24 was calculated according to the RTDP NRC approved methodology described in WCAP-11397-P-A Conservative instrumentation uncertainties were combined with WRB-1 DNBR statistics (from WCAP-14565-P-A) There is a large DNBR margin available prior to plant stretch power uprating

b The retained DNBR margin of 7 46% is sufficient for operation at the current power level (rated core power of 1650 MWth) and the MUR power level (rated core power of 1673 MWth) The DNBR penalties that total 8.40% include the effect of the stretch power uprate The DNBR safety limit will not be further increased to account for the effect of the stretch power uprate Instead, the Design Limit DNBR will be reduced using calculated instrumentation uncertainty from WCAP-15591 Rev. 1, the rod bow penalty will be reduced, and, if necessary, the design FdH limit will be decreased These changes will be addressed in the Stretch Power Uprate License Amendment Request Submittal

26 Section 5 1 of the Westinghouse report indicates that a T_{AVG} range of 556.3°F to 573 0°F was considered in the Non-LOCA transients reanalyzed for the fuel transition and power uprate program. The KNPP draft COLR requires that during steady state power operation T_{AVG} shall be < 576.7°F Assuming a lower temperature in the transient analyses is non-conservative with respect to DNBR Please discuss this discrepancy, including why DNBR limits are not exceeded during Anticipated Operational Occurrences.

NMC Response: With respect to DNBR analyses, Standard Thermal Design Procedure (STDP) events are analyzed at the RTDP nominal T_{AVG} of 573.0°F plus the RTDP uncertainty of 6 0°F. The Revised Thermal Design Procedure (RTDP) events are analyzed at the RTDP nominal T_{AVG} of 573.0°F (with the RTDP uncertainty of 6 0°F incorporated in the RTDP DNBR design limits, see response to RAI A2-1). Both STDP and RTDP DNBR events are therefore analyzed up to 579 0°F.

The COLR T_{AVG} limit of 576 7°F allows for control board indication error, which has been included in the total RTDP T_{AVG} uncertainty. The DNBR limits will not be exceeded as long as the T_{AVG} indicated at the control board is less than 576 7°F.

27 Section 5 1 of the Westinghouse report indicates that up to 10 percent symmetric steam generator tube plugging was considered in the analyses Please provide a discussion of the expected impacts of this level of asymmetric tube plugging on the non-LOCA transient results

NMC Response: Asymmetric steam generator tube plugging is related to the issue of reactor coolant loop flow asymmetry. Westinghouse has conservatively determined via engineering analysis that the maximum loop-to-loop flow imbalance for a two-loop Westinghouse plant with as much as a 10% difference in loop-to-loop steam generator tube plugging levels is 5% of nominal loop flow.

The non-LOCA transients most significantly impacted by loop-to-loop flow asymmetry are those events for which the existence of a loop flow imbalance may directly result in a lower total core flow during the transient relative to the total core flow resulting from an initial symmetric flow condition. These transients are partial loss of forced reactor coolant flow and reactor coolant pump locked rotor. Transients initiated from partial-flow conditions, i e , with less than the total number of reactor coolant pumps in operation, may be similarly affected These transients are uncontrolled RCCA withdrawal from a subcritical condition and RCCA ejection at hot zero power. For a maximum loop-to-loop steam generator tube plugging difference of 10%, the applicable core flow reduction is 1 1% of the nominal core flow. This effect has been explicitly accounted for in the safety analyses for KNPP via a penalty applied to the calculated transient core flow

28 Inadvertent loading and operation of a fuel assembly in an improper position is not listed in Table 5.1.7 as an event that was reanalyzed as part of this amendment request. Please discuss the reason for this and why this event remains acceptable under the proposed fuel transition and uprate power program. Is this event not part of the KNPP licensing basis?

NMC Response This event is not a part of the Kewaunee licensing basis.

29. Tables 5.1-7 and 5.1-1 of the Westinghouse report provide a summary of the methods used and the results of the non-LOCA transient analyses The information provided appears to be for the Westinghouse fuel only, as only the Westinghouse DNB correlations are listed. Please provide the analogous results which demonstrate the acceptability of the Framatome Fuel over the transition operating cycles

NMC Response: The non-fuel related non-LOCA transient analyses results (e.g , Reactor Coolant System pressure, Main Steam System pressure, pressurizer does not become water solid, etc) are applicable to transition cores as well as full 422V+ cores As such, these results are bounding for Framatome ANP fuel

The fuel-related non-LOCA transient analyses results (thermal-hydraulic analyses results) (e.g , DNBR, fuel centerline temperature, peak clad temperature, percent of rods in DNB, etc) for Framatome ANP fuel will be generated with the approved methods of the NMC reload safety evaluation methods topical report for KNPP (Reference 1) The effects of all uprates and the mixed core will be evaluated Consistent with previous cycles, the thermal-hydraulic analyses for the Framatome ANP fuel will be generated during the reload safety evaluation process and will be documented in Reload Safety Evaluation report and in the Stretch Power Uprating Submittal Again, consistent with previous cycles the detailed documentation will be provided in the Updated Safety Analysis Report (USAR), completed six months after start-up

The analytic margin to the DNBR limit for the Framatome ANP fuel can be expected to increase for the fuel transition alone This is due to the fact that the increase in the Cycle 26 T_{ave} will be more than offset by the increase in local flow due to the mixed core effects and the decrease in $F_{\Delta H}$ due to the once-burned status of the Framatome ANP fuel The analytic margin to the DNBR limit for the Framatome ANP fuel can also be expected to increase for the fuel transition and measurement uncertainty recapture (MUR) uprate. This is due to the same reasons as above plus the fact that the MUR uprate relies on exchanging the power increase with a power uncertainty decrease However, the analytic margin to the DNBR limit for the Framatome ANP fuel for the fuel transition and the full stretch uprate can be expected to decrease somewhat This is due to the fact that the power increase will erode slightly more margin than is gained by the net result of the increase in the Cycle 26 T_{ave}, the increase in local flow due to the mixed core effects and the decrease in $F_{\Delta H}$ due to the once-burned status of the Framatome ANP fuel

References

1 U S. Nuclear Regulatory Commission Letter, entitled "Kewaunee Nuclear Power Plant – Review for Kewaunee Reload Safety Evaluation Methods Topical Report WPSRSEM-NP, Revision 3 (TAC No. MB0306)", dated September 10, 2001

30. Regarding the reanalysis of the Uncontrolled RCCA Withdrawal from a Subcritical Condition transient

    a. Please provide the technical basis for the assumptions of an 1100 pcm Doppler power defect, and the change in initial power level from 1 0E-13 to 1 0E-9 Are these conservative assumptions?

    b. Please verify that maximum values of the effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (l*) are used in the analyses

    c Table 5 1-1 provides the DNBR results for the W-3 correlation only, which is valid below the first mixing vane grid. Please provide the MDNBR results for above the first mixing vane grid, which are calculated using the WRB-1 correlation. Also, please provide the basis for the analysis limit of 1 39 for the W-3 correlation.

NMC Response

a The technical basis for the assumption of an 1100 pcm Doppler power defect is as follows. For the Uncontrolled RCCA Withdrawal From A Subcritical Condition event, a low value (in absolute magnitude) of the Doppler-only power coefficient (DPC) produces conservative analysis results by reducing the amount of Doppler feedback and thereby increasing the neutron flux peak. A Doppler-only power defect of 1100 pcm represents a conservatively low value (in absolute magnitude) for the DPC. The conservatism of the Doppler-only power defect is confirmed each reload during the reload safety evaluation by comparing the cycle-specific value to the safety analysis value

Regarding the change in initial power from 1.0E-13 to 1.0E-9, the assumed value is conservatively lower than the power level expected for any zero-power or near critical shutdown condition, and therefore the change is acceptable.

b A maximum delayed neutron fraction of 0.0072 is used in the analysis. As the analysis is performed using a diffusion-theory computer code (TWINKLE), a prompt neutron lifetime is not used explicitly There is, however, a best-estimate "prompt neutron lifetime" that is implicit in the cross-section data and neutron velocities used in the transient simulation Use of a best-estimate "prompt neutron lifetime" is acceptable because the overall conservatism of the analysis is ensured by the moderate sensitivity to this parameter, and by the use of conservative values for other critical parameters (reactivity insertion rate, delayed neutron fractions, reactivity feedback coefficients, reactor trip setpoint, trip reactivity, etc)

c. "Above the first mixing vane grid, the WRB-1 DNB correlation is applicable and the minimum DNBR is 1.879/1 863 (thimble/typical cell), the corresponding DNBR limit is 1.25 in order to maintain the same generic DNBR margin as for the RTDP events Below the first mixing vane grid, the W-3 DNB correlation is applicable and the minimum DNBR is 1.588/1.733 (thimble/typical cell), the corresponding DNBR limit is 1.39 in order to maintain the same generic DNBR margin as for the RTDP events "

31 Regarding the reanalysis of the Uncontrolled RCCA Withdrawal from Power transient:

a. RETRAN (a system code) rather than a subchannel code such as VIPRE is used for DNBR analysis for this transient. The use of the RETRAN DNBR model requires certain user-input values (not listed here because this is shown as proprietary on page 55 of WCAP-14882-P-A) Please discuss how this user-input was determined for KNPP for the fuel transition and power uprate program

b NUREG-0800, Standard Review Plan, Section 15 4.2 lists the acceptance criteria for this event as DNBR>limit and fuel centerline temperature remains less than the melt temperature. Please provide quantitative results, which demonstrate that the fuel centerline temperature acceptance criteria is satisfied

c. The tables of results are shown for three power levels: 100 percent, 60 percent and 10 percent power. What is the 100 percent power value (MWt) assumed in the analyses for minimum DNBR and fuel centerline temperature? Did the analysis include calorimetric uncertainty?

NMC Response:

a. For transients in which the RCS flow remains unchanged, and in which there are no significant radial peaking effects, it is possible to use the core thermal safety limits figure as an indication of how the DNBR changes in response to changes in the core power, average reactor coolant temperature, and pressure. This approach is valid since the core thermal safety limits are linear with changes in these parameters, as shown by the sloped lines at the right side of Technical Specification Figure 2.1-1 The core thermal limits are input to the Westinghouse RETRAN code via a set of partial derivatives that cover any temperature, pressure (from the low pressurizer pressure safety analysis setpoint to the high pressurizer pressure setpoint), and power condition up to a power level of 118%

These partial derivatives are defined such that they correspond to being at the safety limit DNBR at the same condition or before the actual core thermal safety limits; the partial derivatives are exactly parallel, whereas the actual core thermal safety limits are very slightly non-linear. In defining the Westinghouse RETRAN partial derivatives, it is ensured that they are equal to or lower than the core thermal safety limits for all possible temperature, pressure, and power conditions The partial derivatives are most accurate at the safety limit DNBR The partial derivatives are used along with the DNBR at nominal conditions to define how close the DNBR is to the safety limit DNBR, based on changes in the core power, average coolant temperature, and RCS pressure.

The partial derivatives are used in the analyses of selected anticipated operating occurrences (AOOs) The initial conditions for an AOO correspond to nominal full power conditions This nominal point could be plotted on the core thermal safety limits Figure and it would have a corresponding DNBR equal to the DNBR at nominal conditions (typically greater than 2.0) If an AOO were to occur, the DNBR would change based on how the plant conditions changed relative to the core thermal limits (the core thermal limits represent the conditions where the plant is at the safety limit DNBR). If the transient resulted in an increase in power, for instance, the DNBR would decrease from the nominal value and would approach the safety limit DNBR, i.e , the core safety limits, up until just beyond the time of reactor trip The amount that the DNBR changes for a given set of changes in power, temperature and pressure is defined by the partial derivatives and the difference between the nominal DNBR and the safety limit DNBR.

Provided that the RCS flow remains unchanged, the relationship between T_{inlet}, reactor coolant system pressure, and power (thermal power) is fairly linear for a constant DNBR, as exhibited by the core thermal DNB limits, which are based on VIPRE code calculations at the DNBR safety limit As long as the RCS flow remains constant, and there is no asymmetric peaking effects for a given transient, the partial derivatives of the core thermal DNB limits can be used to determine the DNBR trend for transient changes in the reactor coolant temperature and pressure, and core power. Westinghouse has employed this approach of using the partial derivatives of the core thermal DNB limits since the 1970s for all Westinghouse-designed PWRs

b During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak KW/ft fuel rods will not exceed the uranium dioxide melting temperature at the 95 percent confidence level. In general, maximum fuel centerline temperatures occur near beginning-of-life (BOL) and near-BOL analyses of the fuel will provide the most restrictive local power limit that meets the fuel no-melt design criteria. By precluding uranium dioxide melting, fuel geometry is preserved and eliminates possible adverse effects of molten uranium dioxide on the cladding The melting temperature of uranium dioxide is taken as 5080 °F (Reference 1) unirradiated and decreasing 58 °F per 10,000 MWD/MTU. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700 °F has been calculated as over power limit. This provides sufficient margin for uncertainties in the thermal evaluation The fuel centerline temperature acceptance criterion is satisfied by demonstrating that the peak linear power resulting from overpower trainsients/operator errors (assuming a maximum overpower of 118 percent) does not exceed 22 54 KW/ft for Westinghouse 422V+ fuel The 118% power limit is used in conjunction with the nominal linear heat rate (kW/ft) and the transient-specific Fq to calculate a peak linear heat rate (kW/ft) value, which is then compared to the kW/ft limit that corresponds to fuel centerline melting

The transient-specific Fq is determined via a detailed core analysis of RCCA bank withdrawals from different power levels and times in core life The peak heat flux calculated in the Uncontrolled RCCA Withdrawal at Power analysis is 117%, which is less than the prescribed limit

Reference

1 Christensen, J A , Allio, R. J. and Biancheria, A , Melting Point of Irradiated UO2", WCAP-6065, February 1965.

c. As the revised thermal design procedure (RTDP) was employed, the nominal core power of 1772 MWt was assumed in the analysis The calorimetric uncertainty is accounted for in both the DNBR safety analysis limit and the heat flux limit

32 Regarding the reanalysis of the RCCA Misalignment transient

a NUREG-0800, Standard Review Plan, Section 15.4.3 lists the acceptance criteria for this event as DNBR>limit and fuel centerline temperature remains less than the melt temperature. Please provide quantitative results which demonstrate that these acceptance criteria are satisfied for each of the cases analyzed.

b Please list and provide a reference to the nuclear physics computer codes used to analyze the steady-state power distributions.

c. Please provide the maximum value of the resulting radial power peaking factor ($F^N_{\Delta H}$). When in core life does this value occur?

d. What power level (MWt) is assumed in the analyses for minimum DNBR and fuel centerline temperature? Did the analysis include calorimetric uncertainty?

NMC Response: In the context of RCCA Misalignment, two categories are defined These RCCA malfunctions are treated as Condition II events As condition II events, these must be shown to meet the DNB design basis.

a. For the static misaligned rod case, the peak $F^N_{\Delta H}$ that will result in a minimum DNBR equal to the Safety Analysis (SAL) DNBR (1.34) is calculated by the VIPRE code Acceptability is determined by the nuclear designer by showing that the calculated $F^N_{\Delta H}$ in the misaligned configuration is less than the $F^N_{\Delta H}$ limit. Calculations performed using the core models discussed in Section 3 4, of attachment 4 to KNPP LAR 187, resulted in a maximum $F^N_{\Delta H}$ (including uncertainty) of 1.827 which is less than the limit of 2 03

   For the dynamic dropped rod case the VIPRE code is used to determine power levels for a range of inlet temperatures and core pressures such that the minimum calculated DNBR equals the safety analysis DNBR of 1.34 The VIPRE runs result in a set of power levels as a function of inlet temperature and pressure which are called dropped rod limit lines (DRLL) Transient RCS statepoints (temperature, pressure, and power), that cover a range of reactivity insertion mechanisms, and DRLL are used by Nuclear Design analysis in accordance with approved methodology (see response to RAI question # 35).

   This method confirms that the pre-transient $F^N_{\Delta H}$ required for DNB to occur during the transient is greater than the $F^N_{\Delta H}$ design limit of 1 574(1.70/1.08) Since results are sensitive to MTC, cases with core average temperatures of 577 and 570 deg-F were evaluated Pre-transient $F^N_{\Delta H}$'s were 1 584 and 1.566 respectively While the latter represents a small violation of about 0 5%, the criteria was satisfied by applying available DNB margin Therefore, the transient RCS setpoints do not violate the DNBR design criteria

   For the dynamic dropped rod case, fuel centerline melting criteria is satisfied by showing the maximum Fq during the transient (2.046 calculated for RTSR) is less than Fq limit for centerline melt (2 59 in this case, which corresponds to 22 54 KW/ft limit discussed in response to RAI question # 31).

b. The ANC code (References 27 & 28) was used to calculate steady-state power distributions

c The maximum $F^N_{\Delta H}$ for the static misaligned rod was 1 827(see 1. above). This occurred at a cycle burnup near mid-cycle of 8000 MWD/MTU. Calculations are done at the cycle burnup where the peak steady state $F^N_{\Delta H}$ occurs (see Figure 3-7)

For the static misaligned rod case, the peak FdH limit that results in the minimum DNBR equal to the SAL DNBR of 1 34 was calculated by the VIPRE code at the nominal operating condition described in table 4-1 of the LAR 1897 submittal, attachment 4, with a DNB-limiting axial power shape Applicability of the DNB-limiting shape is verified on a cycle-specific basis

To address the small violation of the FdH for the dropped rod case, retained DNBR margin between the SAL DNBR and Design Limit (DL) DNBR was applied. The required DNBR margin for the 0 5% FdH change is much less than the available margin of 7 46% listed in Table 4-4 of the LAR 1897 submittal, attachment 4 Since the FdH values of the dropped rod event are verified for every reload cycle, the allocation of the DNBR margin is evaluated on a cycle-specific basis. As an example, the KNPP Cycle 26 reload design does not violate the dropped rod limit lines and therefore does not require any allocation of the DNBR margin.

d. Core power level of 1772 MWT was used for these evaluations Calorimetric uncertainty of 2 0 percent is included

33 Regarding the reanalysis of the Chemical and Volume Control System Malfunction:

a The maximum dilution flow rate used in the analysis was decreased from 180 gpm to 80 gpm The KNPP USAR states that 80 gpm is the maximum dilution flow with two charging pumps operating and three letdown orifices in service Please provide the technical basis for reducing the dilution flow assumption. Is there ever a situation in which all three charging pumps can be operating and contributing to the dilution flow? Demonstrate that the 80 gpm assumption is conservative

NMC Response: All important safety analysis assumptions must conservatively bound the operation of the plant Consideration must also be given to a single active failure in the most limiting system. The conservatively bounding maximum capacity of each of the three charging pumps is 60 gpm (see KNPP USAR Section 14.1 4) The normal operating state is to have two charging pumps operating; one in manual control and one in automatic control, and one pump not operating An initiating event is assumed that results in a maximum of two pumps delivering the maximum flow for a total dilution rate of 120 gpm.

The failure causing the event initiation can be postulated to be in the control system of the pump in automatic control which combined with the pump in manual control in the worst-case initial state (full flow) would provide the 120 gpm Note that the pump that is not operating would not, absent an additional active failure, provide additional flow. Note that the charging pumps do not have an automatic start feature so that a failure that results in the starting of the pump is not very credible.

There is no credible failure that can result in both the pump that is in automatic control and the pump that is not operating delivering maximum flow such that combined with the pump in manual control in the worst-case initial state (full flow) could provide a dilution flow equivalent to all three charging pumps at maximum flow (180 gpm). Although it is theoretically possible to have all three charging pumps providing dilution flow, it precluded by procedural administrative control and training and it is not within the scope of safety analysis to specifically consider sabotage or gross operator negligence

Based on the above discussion, the CVCS Malfunction analysis in LAR 187 is being retracted and will be superseded with a re-analysis submitted with the formal RAI responses (see part d below) that assumes a maximum dilution flow of 120 gpm.

b. The initial boron concentration values have been changed in the analyses During refueling, the value increased from 2200 to 2250 ppm. During Startup, the value decreased from 2200 to 1800 ppm. During power operation, the value increased from 1600 to 1780 ppm Please provide the technical basis for these changes

NMC Response: The initial boron concentrations are based on ANC calculations for Kewaunee using the core physics models discussed in Section 3.4 of attachment 4 of KNPP's LAR 187 submittal. The refueling calculations involve the use of "cold" models explicitly developed for low temperature calculations Boron values were selected to bound all three cycles discussed in Section 3 4.

c NUREG-0800, Standard Review Plan, Section 15 4.6 also lists DNBR and primary and secondary system pressure acceptance criteria for this event. Please discuss the analyses performed and provide results which verify that these acceptance criteria are satisfied

NMC Response. The analysis was performed to determine the amount of time available for operator or automatic mitigation of the boron dilution event prior to the complete loss of plant shutdown margin. As the calculated times are accepted to be sufficient, it can be assumed that automatic or manual actions will effectively prevent the complete loss of minimum allowable shutdown margin. With no loss of shutdown margin, the condition of the plant at any point in the transient is within the bounds of those calculated for other USAR Condition II transients Therefore, as long as the Condition II criteria are shown to be met for the balance of the USAR Condition II transients, it can be concluded that the same criteria are met for the USAR boron dilution transient (Chemical and Volume Control System Malfunction)

d. Please provide the calculated available operator action times (time between alarm and loss of shutdown margin) for the three modes of operation analyzed How did the operator action time change from the previous analysis of record?

NMC Response: The CVCS Malfunction transient was reanalyzed with a maximum dilution flow of 120 gpm (see Section 5 1.4 of Attachment B) The calculated available operator action times for the three modes of operation are as follows

Mode		COMPLETE SHUTDOWN MARGIN LOST **(minutes)**		
		Cycle 25	**Cycle 26**	**Limit**
Refueling		> 30 0	31 6	> 30 0
Startup		> 15 0	28 8	> 15.0
At Power	Auto	N/A	25 1	> 15.0
	Manual	N/A	22 7*	> 15.0

* Time available after reactor trip

The limit values are consistent with the criteria outlined in Section 15 4 6 of the Standard Review Plan (SRP) dated September 1975. Note that the current KNPP licensing basis does not require specific criteria for operator action times for the At Power events. For the Cycle 25 At Power - Auto event, loss of 1% shutdown margin (not loss of complete shutdown margin) occurs 10 4 minutes after event initiation For the Cycle 25 At Power – Manual event, loss of complete shutdown margin after reactor trip occurs 9 4 minutes after reactor trip. These available response times are adequate to ensure that all Condition II acceptance criteria are met It is also noted that for Cycle 26 and beyond, consistent with SRP 15.4.6 the full set of criteria listed above will be used as the licensing basis for KNPP.

34. Regarding the reanalysis of the RCCA Ejection event:
    - a NUREG-0800, Standard Review Plan, Section 15 4 8 and Regulatory Guide 1.77 also list reactor pressure as an acceptance criteria for this event. Please discuss the analyses performed and provide results which verify that this acceptance criteria are satisfied
    - b Proposed USAR Section 14.2 6, insert B states that a minimum value of the Doppler power defect and moderator feedback is assumed in the analyses. Does this imply an absolute value, such that a least negative value for these parameters is used?
    - c. USAR Table 14 2.6-1 shows the maximum cladding average temperature for the EOL-HZP case to be 2987 °F The analysis limit according to the Westinghouse methodology is 2700 °F. Based on this, please explain why this is acceptable? Is cladding failure assumed in this case?
    - d Please discuss how the Doppler feedback reactivity weighting factors are calculated and applied It would be helpful if this could be demonstrated through a simple sample calculation Please provide a reference to the NRC-approved methodology for these factors.

NMC Response:

- a. With respect to the reactor coolant system pressure, the RCCA ejection transient has been analyzed on a generic basis for Westinghouse-designed PWRs as detailed in WCAP-7588, Revision 1-A (January 1975).
- b. Yes it does imply absolute values, and least negative values are used.
- c. As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (W. J Johnson to R C Jones, dated October 23, 1989), the 2700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long term core coolability and compliance with applicable offsite dose requirements are those defined in Insert A of USAR Section 14 2 6 (fuel pellet enthalpy, RCS pressure, and fuel melt)

d. The one-dimensional TWINKLE steady-state calculation is tuned for each core design to yield a conservative, bounding (small absolute value) zero-to-full power Doppler defect compared to the actual cycle design value for that plant. Doppler weighting factors (DWF) are then applied during the transient calculation to give a conservative representation of the increase in Doppler feedback due to the increased radial and axial peaking factors resulting from the ejected rod. The Doppler weighting factors used were obtained from static one-dimensional axial and radial calculations as a function of axial and radial peaking factors, and have been normalized to give no weighting (DWF_{Radial} and DWF_{Axial} = 1.0) at the design peaking factor The total DWF is given by the equation: $DWF_{Total} = DWF_{Radial} \times DWF_{Axial}$ and is a multiplier on the Doppler defect.

As a typical example, for a HZP BOL case with a conservative F_q of 11, $F_{\Delta H}$ of 5 5, and an F_z of 1.8, the total Doppler weighting factor is:

$$DWF_{Total} = 1\ 842 \times 1.09 = 2.008.$$

This method of calculating the Doppler weighting factors is the same as that used to support the Rod Ejection topical report (WCAP-7588 Rev. 1-A), which was approved by the NRC in 1973 The conservatism of this approach has been demonstrated by comparison to a three-dimensional transient analysis shown in the same report, as well as by a similar comparison shown in the Reference listed below, which is currently under NRC review

Reference

1 C L Beard et. al , "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics", WCAP-15806-P (Proprietary) and WCAP-15807-NP (Non-Proprietary).

35 Please expand Table 5 1 7 of the submittal to include the following information regarding the computer codes and methodologies used in the new analyses in a tabulate form, include ANC and PHOENIX· 1) the computer codes and methodologies used in each of the transient and accident analyses, 2) the staff review and approval status of these codes and methodologies, 3) the conditions and restrictions for each of the code and methodologies, 4) how these conditions and restrictions are satisfied in each application to the new analyses, and 5) is the use of these codes and methodologies in the new analyses consistent with the current licensing basis

NMC Response:

1 The computer codes and methodologies used in each of the non-LOCA transient analyses are listed in Table RAI35-1 included below

2 As indicated by Tables RAI35-2 through RAI35-6 and Tables RAI35-9 and RAI35-10, the NRC staff has approved all codes that were used in the non-LOCA transient analyses for Kewaunee. As for the applicable non-LOCA transient analysis methodologies, these have been reviewed and approved by the NRC staff via transient-specific topical reports (WCAPs) and/or through the review and approval of plant-specific safety analysis reports (see Table RAI35-1)

3 Code and methodology restrictions are specified in applicable SERs Tables RAI35-2 through RAI35-6 and Tables RAI35-9 and RAI35-10 identify the SER conditions and restrictions for each of the computer codes listed in Table RAI35-1. Similarly, Tables RAI35-7 and RAI35-8 identify the SER conditions and restrictions for each methodology that has an approved topical report associated with it.

4. Tables RAI35-2 through RAI35-10 also provide the justifications for how each SER condition/restriction is satisfied in the Kewaunee analyses. To help ensure that all applicable SER conditions and restrictions are satisfied for each transient analysis that is performed, Westinghouse utilizes internal methodology guidance documents Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development, and a discussion of the expected transient analysis results.

5 Although different codes and methods were applied in the new analyses, the application of these codes and methods to the KNPP licensing basis is valid (i e , all restrictions and limitations of methodologies have been met). Furthermore, a comparison of the results (see USAR markups) indicates that the use of the codes and methods are consistent with the current licensing basis results.

Table RAI35-1:	Computer Codes and Methodologies Used in Non-LOCA Transient Analyses for Kewaunee		
USAR Section	**Event Description**	**Applicable Code(s)**	**Applicable Methodology**
14 1 1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE (WCAP-7979-P-A), FACTRAN (WCAP-7908-A), VIPRE (WCAP-14565-P-A)	SAR submittals
14.1 2	Uncontrolled RCCA Withdrawal at Power	RETRAN (WCAP-14882-P-A)	SAR submittals
14.1.3	RCCA Misalignment	LOFTRAN (WCAP-7907-P-A), VIPRE (WCAP-14565-P-A) ANC (WCAP-10965-P-A) PHOENIX-P (WCAP-11596-P-A)	WCAP-11394-P-A
14 1.4	Chemical and Volume Control System Malfunction	N/A	SAR submittals
14 1.5	Startup of an Inactive Reactor Coolant Loop	N/A	Event precluded by Tech Specs
14 1.6	Feedwater Temperature Reduction Incident	N/A	SAR submittals
14.1 6	Excessive Heat Removal Due to Feedwater System Malfunctions	RETRAN (WCAP-14882-P-A), VIPRE (WCAP-14565-P-A)	SAR submittals
14 1 7	Excessive Load Increase Incident	N/A	SAR submittals
14.1.8	Loss of Reactor Coolant Flow	RETRAN (WCAP-14882-P-A), VIPRE (WCAP-14565-P-A)	SAR submittals
14.1.8	Locked Rotor	RETRAN (WCAP-14882-P-A), VIPRE (WCAP-14565-P-A), FACTRAN (WCAP-7908-A)	SAR submittals
14.1.9	Loss of External Electrical Load	RETRAN (WCAP-14882-P-A)	SAR submittals
14 1.10	Loss of Normal Feedwater	RETRAN (WCAP-14882-P-A)	SAR submittals
14 1.11	Anticipated Transients Without Scram	N/A	N/A
14 1 12	Loss of AC Power to the Plant Auxiliaries	RETRAN (WCAP-14882-P-A)	SAR submittals
14 2 5	Steam Line Break	RETRAN (WCAP-14882-P-A), VIPRE (WCAP-14565-P-A) ANC (WCAP-10965-P-A) PHOENIX-P (WCAP-11596-P-A)	SAR submittals
14 2 6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	TWINKLE (WCAP-7979-P-A), FACTRAN (WCAP-7908-A)	WCAP-7588, Rev 1-A

Table RAI35-2: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN	
Computer Code:	RETRAN
Licensing Topical Report:	WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
Date of NRC Acceptance:	February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***
	Justification The transients listed in Table 1 of the SER are: *Feedwater system malfunctions,* *Excessive increase in steam flow,* *Inadvertent opening of a steam generator relief or safety valve,* *Steam line break,* *Loss of external load/turbine trip,* *Loss of offsite power,* *Loss of normal feedwater flow,* *Feedwater line rupture,* *Loss of forced reactor coolant flow,* *Locked reactor coolant pump rotor/sheared shaft,* *Control rod cluster withdrawal at power,* *Dropped control rod cluster/dropped control bank,* *Inadvertent increase in coolant inventory,* *Inadvertent opening of a pressurizer relief or safety valve,* *Steam generator tube rupture.* The transients analyzed for Kewaunee using RETRAN are: *Uncontrolled RCCA withdrawal at power (USAR 14.1.1),* *Excessive heat removal due to feedwater system malfunctions (USAR 14.1.6),* *Loss of reactor coolant flow (USAR 14.1.8),* *Locked rotor (USAR 14.1.8),* *Loss of external electrical load (USAR 14.1 9),* *Loss of normal feedwater (USAR 14.1 10),* *Loss of AC power to the plant auxiliaries (USAR 14.1.12),* *Steam line break (USAR 14.2 5).*

Table RAI35-2: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN	
Computer Code	RETRAN
Licensing Topical Report:	WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
Date of NRC Acceptance:	February 11, 1999 (SER from F Akstulewicz (NRC) to H Sepp (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
2.	***" WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."***
	Justification The Kewaunee Nuclear Power Plant is a 2-loop Westinghouse-designed plant that was "currently operating" at the time the SER was written (February 11, 1999) Therefore, additional justification is not required.
3.	***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."***
	Justification The input data used in the RETRAN analyses performed by Westinghouse came from both NMC and Westinghouse sources. Assurance that the RETRAN input data is conservative for Kewaunee is provided via Westinghouse's use of transient-specific analysis guidance documents Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results Based on the analysis guidance documents, conservative, plant-specific input values were requested and collected from the responsible NMC and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A, the safety analysis input values used in the Kewaunee analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Table RAI35-3: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - TWINKLE	
Computer Code	TWINKLE
Licensing Topical Report:	WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975
Date of NRC Acceptance:	July 29, 1974 (SER from D. B. Vassallo (U S Atomic Energy Commission) to R. Salvatori (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
	There are no conditions, restrictions, or limitations cited in the TWINKLE SER.
	Justification As the TWINKLE SER does not cite any conditions, restrictions, or limitations, additional justification is not required.

Table RAI35-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN	
Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."***
	Justification The FACTRAN code was used in the analyses of the following transients for Kewaunee: Uncontrolled RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), and RCCA Ejection (USAR 14.2.6). Initial fuel temperatures were used as FACTRAN input in the Locked Rotor and RCCA Ejection analyses. The assumed fuel temperatures for these transients were calculated using the NRC-approved PAD 4.0 computer code (see WCAP-15063-P-A). As indicated in WCAP-15063-P-A, the NRC has approved the method of determining uncertainties for PAD 4.0 fuel temperatures.
2.	***"Table 2 presents the guidelines used to select initial temperatures."***
	Justification In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the Locked Rotor and RCCA Ejection analyses for Kewaunee. The assumed fuel temperatures, which were calculated using the PAD 4.0 computer code (see WCAP-15063-P-A), include uncertainties and are conservatively high.
3.	***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."***
	Justification The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the Locked Rotor and RCCA Ejection transients, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.

Table RAI35-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN	
Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
4.	***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."***
	Justification Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), RCCA Ejection (USAR 14.2.6)). Therefore, additional justification is not required.
5.	***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."***
	Justification At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), RCCA Ejection (USAR 14.2.6)).
6.	***"Although time-<u>independent</u> mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-<u>dependent</u> mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."***

Table RAI35-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN	
Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
	Justification The three transients that were analyzed with FACTRAN for Kewaunee (RCCA Withdrawal from a Subcritical Condition (USAR 14.1.1), Locked Rotor (USAR 14.1.8), and RCCA Ejection (14.2.6)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the Locked Rotor and RCCA Ejection transients, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.
7.	***"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."***
	Justification The condition of concern (T(centerline) – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Kewaunee.

Table RAI35-5: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - LOFTRAN	
Computer Code:	LOFTRAN
Licensing Topical Report:	WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.
Date of NRC Acceptance:	July 29, 1983 (SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:*** - ***Feedwater System Malfunction*** - ***Excessive Increase in Steam Flow*** - ***Inadvertent Opening of a Steam Generator Relief or Safety Valve*** - ***Steamline Break*** - ***Loss of External Load*** - ***Loss of Offsite Power*** - ***Loss of Normal Feedwater*** - ***Feedwater Line Rupture*** - ***Loss of Forced Reactor Coolant Flow*** - ***Locked Pump Rotor*** - ***Rod Withdrawal at Power*** - ***Rod Drop*** - ***Startup of an Inactive Pump*** - ***Inadvertent ECCS Actuation*** - ***Inadvertent Opening of a Pressurizer Relief or Safety Valve*** ***This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."***
	Justification The LOFTRAN code was only used in the analysis of the Rod Drop transient (USAR 14.1.3) for Kewaunee. As this transient matches one of the transients listed in the SER, additional justification is not required.

Table RAI35-6: Approval Status & SER Requirements for Core Analysis Codes - VIPRE	
Computer Code:	VIPRE
Licensing Topical Report:	WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999.
Date of NRC Acceptance:	January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."***
	Justification The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Kewaunee 422V+ fuel. The validity of the WRB-1 DNB correlation applicability to the 422V+ fuel was discussed in the response to Attachment 3 RAI No. 21. The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Kewaunee 422V+ fuel are the same as those previously used and approved for the safety evaluations of other Westinghouse two-loop plants using the same fuel design.
2.	***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."***
	Justification The core boundary conditions for the VIPRE calculations for the 422V+ fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the RTSR. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology WCAP-9272-P-A.
3.	***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."***
	Justification As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used for the DNB analyses of 422V+ fuel in Kewaunee. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45 if 500 < P < 1000).

Table RAI35-6: Approval Status & SER Requirements for Core Analysis Codes - VIPRE	
Computer Code:	VIPRE
Licensing Topical Report:	WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999.
Date of NRC Acceptance:	January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
4.	***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extent to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***
	Justification The application of the VIPRE to the 422V+ fuel upgrade in Kewaunee did not include usage in the post-CHF region.

Table RAI35-7: Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – Dropped Rod	
Transient:	RCCA Misalignment (Dropped Rod)
Licensing Topical Report:	WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.
Date of NRC Acceptance:	October 23, 1989 (SER from A. C. Thadani (NRC) to R. A. Newton (WOG))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"The Westinghouse analysis, results and comparisons are reactor and cycle specific. No credit is taken for any direct reactor trip due to dropped RCCA(s). Also, the analysis assumes no automatic power reduction features are actuated by the dropped RCCA(s). A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."***
	Justification For the reference cycle assumed in the Kewaunee 422V+ fuel transition/uprate program, it is affirmed that the methodology described in WCAP-11394-P-A was performed and the required comparisons have been made with acceptable results (DNB limits are not exceeded).

Table RAI35-8: Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – RCCA Ejection	
Transient:	RCCA Ejection
Licensing Topical Report:	WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.
Date of NRC Acceptance:	August 28, 1973 (SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
1.	***"The staff position, as well as that of the reactor vendors over the last several years, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection accident to 280 cal/gm. This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO_2 have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Westinghouse has decreased their limiting fuel failure criterion from 280 cal/gm (somewhat less than the threshold of significant conversion of the fuel thermal energy to mechanical energy) to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision on the side of safety, the staff concludes that it is an acceptable fuel failure criterion."***
	Justification **The maximum fuel pellet enthalpy at the hot spot calculated for each Kewaunee-specific RCCA Ejection case is less than 200 cal/gm. These results satisfy the fuel failure criterion accepted by the staff.**
2.	***"Westinghouse proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation imposed in the AEC ECCS Interim Acceptance Criteria, this is felt to be adequate in view of the relatively short time at temperature and the highly localized effect of a reactivity transient."***
	Justification As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (W. J. Johnson to R. C. Jones, dated October 23, 1989), the 2700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long term core coolability and compliance with applicable offsite dose requirements are those defined in the suggested revisions to KNPP USAR Section 14.2.6 (fuel pellet enthalpy, RCS pressure, and fuel melt).

Table RAI35-9: Approval Status & SER Requirements for Core Analysis Codes - ANC	
Computer Code:	ANC
Licensing Topical Report:	WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code" September 1986.
Date of NRC Acceptance:	June 23, 1986 (SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
	Although there are no conditions, restrictions, or limitations explicitly cited in the ANC SER, the SER does conclude that "the ANC code provides an accurate calculation of core reactivity, reactivity coefficients critical boron, rod worths and core power distribution for use in design and safety analyses."
	Justification In support of the Kewaunee fuel transition, the ANC code was used to calculate power distributions for normal (design) and off-normal (safety analysis) conditions, and was also used for reactivity calculations. As these code applications are consistent with those listed in the SER, additional justification is not required.

Table RAI35-10: Approval Status & SER Requirements for Core Analysis Codes – PHOENIX-P	
Computer Code:	PHOENIX-P
Licensing Topical Report:	WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" June 1988.
Date of NRC Acceptance:	May 17, 1988 (SER from A. C. Thadani (NRC) to R. W. Johanson (Westinghouse))
Safety Evaluation Report (SER) Conditions & Justification for the Kewaunee Nuclear Power Plant	
	Although there are no conditions, restrictions, or limitations explicitly cited in the PHOENIX-P/ANC SER, the SER does conclude that "The PHOENIX-P lattice physics methods and the PHOENIX-P/ANC nodal analysis systems described in this report are thus acceptable for use in PWR design analysis."
	Justification As Kewaunee is a PWR plant, it is acceptable to use the PHOENIX-P and PHOENIX-P/ANC code system, and additional justification is not required.

36. Please expand Table 5.1.1 of the submittal to include the following information regarding the assumptions, and results of the new analyses of the new analyses in a tabulate form: 1) initial plant conditions consistent with that operated at uprated power level, 2) other major assumptions used in the new analyses including the assumed limiting single failure and loss of offsite power during the event, 3) state the acceptance criteria applied to each of event analyzed, 4) identify any assumptions and acceptance criteria used in the new analyses that are deviate from the current licensing basis and provide justification for the differences, and 5) provide the results of the new analyses for each event to demonstrate that the consequences of each event analyzed met their acceptance criteria with sufficient margin.

NMC Response:

1. See Attachment B.
2. See Attachment B and Table below.

USAR Section	Event Description	Limiting Single Failure Assumption	Loss of offsite power assumed?
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	One train of protection	No
14.1.2	Uncontrolled RCCA Withdrawal at Power	One train of protection	No
14.1.3	RCCA Misalignment	One train of protection	No
14.1.4	Chemical and Volume Control System Malfunction	One train of protection	No
14.1.5	Startup of an Inactive Reactor Coolant Loop	Event precluded by Tech Specs	
14.1.6	Feedwater Temperature Reduction Incident	(1)	No
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions	One train of protection	No
14.1.7	Excessive Load Increase Incident	(1)	No
14.1.8	Loss of Reactor Coolant Flow	One train of protection	No
14.1.8	Locked Rotor	One train of protection	Yes(2)
14.1.9	Loss of External Electrical Load	One train of protection	No
14.1.10	Loss of Normal Feedwater	One auxiliary feedwater pump	No(3)
14.1.11	Anticipated Transients Without Scram	(4)	(4)
14.1.12	Loss of AC Power to the Plant	One auxiliary feedwater	Yes

USAR Section	Event Description	Limiting Single Failure Assumption	Loss of offsite power assumed?
	Auxiliaries	pump	
14.2.5	Steam Line Break	One safety injection train	Yes
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	One train of protection	No

(1)No protection action is required.
(2)The unaffected reactor coolant pump is assumed to lose power instantaneously at the time of reactor trip.
(3)A loss of normal feedwater transient with a loss of offsite power is assumed in the analysis of USAR
Section 14.1.12.
(4)Transient not reanalyzed.

3. See Attachment B.
4. The USAR markups in Attachment 5 of LAR 187 can be used to identify differences in assumptions and acceptance criteria between the new analyses and the current licensing basis analyses. The differences are attributed to one or more of the following: power uprate, fuel transition, or methodology transition. With respect to acceptance criteria, the only changes that have been made are associated with the Locked Rotor and RCCA Ejection transients. For Locked Rotor, the allowable percentage of fuel rods experiencing DNB was increased from 40% to 50%; justification for this change is provided in the response to Attachment 3 RAI No. 43. For RCCA Ejection, the peak clad temperature limit of 2700°F was deleted and a fuel melt criterion (Fuel Melt must be limited to less than the innermost 10% of the fuel pellet at the hot spot) was added; justification for deleting the 2700°F limit is provided in the response to Attachment 3 RAI No. 34, Part c.
5. See Attachment B.

Due to the early reactor trip that occurs following a locked rotor transient, the secondary side pressurization is less limiting than that calculated for the loss of load transient analysis. For KNPP, the locked rotor RCS pressure limit is equal to 110 percent of the design value, or 2750 psia. For secondary side, the locked-rotor pressure limit is also assumed to be equal to 110 percent of design pressure, or 1210 psia. Since loss-of-load analysis bounds the locked rotor, a specific MSS overpressurization analysis is not performed.

The loss of load/turbine trip (LOL/TT) transient bounds the locked rotor (LR) transient with respect to secondary-side pressurization because the LOL/TT transient results in a power mismatch between the primary and secondary sides. In the LOL/TT analysis, following the loss of secondary load (and assuming a coincident feedwater isolation), the primary-side operates at full power (102%) until rod motion occurs (due to OTDT) at 12.2 seconds. This contrasts with the LR analysis in which rod motion (due to low RCS flow) occurs at 0.8 seconds with a coincident turbine trip/feedwater isolation. Having a period of primary-to-secondary power mismatch is conservative with respect to secondary-side pressurization.

The SLB case without offsite power is the same as the case with offsite power except that the reactor coolant pumps are assumed to coast down as a result of the loss of offsite power. The case without offsite power is historically less limiting than the case with offsite power because the core cooldown is retarded by the reduced reactor coolant flow rate. A comparison of the calculated peak heat flux values for the two cases (28.8%) (with offsite power) and 9.6% (without offsite power)) confirms that the core cooldown is significantly more severe in the case with offsite power. Therefore, a minimum DNBR value is not explicitly calculated for the case without offsite power.

37. Please provide a tabulation to compare the plant operational parameters for the current rated power level and the uprated power level.

NMC Response:

Parameter	1650 MWt (Current w/blowdown)	1650 MWt (BE RTSR)	1673 MWt (BE 1.4% Uprate)	1772 MWt (BE 7.4% Uprate)
Primary Side Parameters				
Core Power (MWt)	1650	1650	1673	1772
NSSS Power (MWt)	1657.1	1657.1	1680.1	1779.1
RCS Flow Rate (gpm/loop)	99000	99000	99000	99000
RCS Pressure (psia)	2250	2250	2250	2250
Vessel Inlet Temp (°F)	533.1	543.4	543.1	541.4
Primary Avg. Temp (Tavg-°F)	562	572	572	572
Vessel Outlet Temp (°F)	590.9	600.6	600.9	602.6
Secondary Side Parameters				
Feedwater Flow Rate (10^6 lbs/hr/SG)	3.592	3.594	3.650	3.892
Feedwater Temp (°F)	430.9	429.9	431.2	436.6
Steam Flow Rate (10^6 lbs/hr)	3.576	3.579	3.634	3.876
Steam Outlet Pressure at SG nozzle (psia)	749	824	819	797
Steam Outlet Pressure at transmitter (psia)	743.6	816.1	810.8	787.4
Steam Outlet Temp (°F)	510.8	521.6	521.0	517.8
Blowdown Flow Rate (gpm)	40	40	40	40
Moisture Carryover (wt %)	0.1	0.1	0.1	0.15
Turbine First Stage Pressure (psia)	552.3	546.9	556.5	598.2
Electrical Parameters				
Electrical Output, Gross (MWe)	556	560.0	566.8	595.1
Electrical Output, Net (MWe)	531	535.0	541.8	569.1

38. Provide technical basis to justify that the safety analyses performed at an uprated power level bound the plant operation at the current rated power level. Please address the affect of different initiate conditions to the analyses.

NMC Response: If rated power level is the only difference between analyses the following holds. In safety analyses that are full-power limited, essentially by definition the higher rated power level analyses bound lower rated power level analyses. In safety analyses that are zero-power limited, higher rated power level analyses and lower rated power level analyses are essentially equivalently bounding. In safety analyses that are intermediate-power limited, a spectrum of intermediate powers are analyzed to capture the limiting scenario caused by the differences in plant state and allowed system availability so that the higher rated power level analyses bound lower rated power level analyses.

The following supports the preceding statements. Rated power level affects power-normalized RPS/ESF setpoints (e.g. a reactor trip at 25% power is a trip at a higher absolute power level for a higher rated power) and power level dependent allowed system availability (e.g., allowed operation with a single reactor coolant pump at 2% rated power allows operation with a single reactor coolant pump at a higher absolute power level for a higher rated power). This implies that higher rated power results in a higher absolute power level at the time of actuation or the allowed power level. Higher rated power level analyses are therefore more limiting.

Since rated power level also affects at power initial conditions additional justification is required. All important safety analyses input assumptions must bound the actual plant. All of the uprate safety analysis input assumptions bound the operation of KNPP at both the non-uprated and uprated power conditions. The table below shows the boundedness of the uprate DNBR analysis.

Parameter	**1650 MWt (BE RTSR)**	**1673 MWt (BE 1.4% Uprate)**	**1772 MWt (BE 7.4% Uprate)**	**Uprate Safety Analysis (includes uncertainties)**
Core Power (MWt)	1650	1673	1772	1807
NSSS Power (MWt)	1657.1	1680.1	1779.1	1815
RCS Flow Rate (gpm/loop)	99000	99000	99000	89,000
RCS Pressure (psia)	2250	2250	2250	2200
Vessel Inlet Temp (°F)	543.4	543.1	541.4	545.2
Primary Avg. Temp (Tavg-°F)	572	572	572	579.0
Vessel Outlet Temp (°F)	600.6	600.9	602.6	612.8

Hence, there is a high confidence that if the uprate safety analysis were to be re-performed with a lower rated power level, all margins to safety analysis acceptance criteria would either be increased or remain unchanged. Therefore, the uprate safety analyses bound lower rated power operation.

39. Proposed UFSAR Table 14.0-2 reflect that there is a 2 percent calorimetric uncertainties considered in the assumed power level in all the safety analyses not associated with DNB calculation using the Westinghouse Revised Thermal Design Procedure (RTDP). However, this fact is not clearly indicated in other discussions of the submittal. Please confirm that this the new analyses to support the proposed fuel change and future power uprate is based on an assumed core thermal power of 1807.5 MWt.

NMC Response: See Table 5.1-7 of Attachment B. The future stretch power uprate license amendment request will support a rated core power of 1772 MWth with a power uncertainty of 0.6%, even though most of the non-LOCA analyses were conservatively performed assuming a 2% power uncertainty, which equates to an assumed core thermal power of about 1807 MWt.

40. It is indicated in the revised UFSAR that TS 2.1.2, 3.10m and the TS Basis for TS 2.2, 3.4, 3.10k, 3.10.l, 3.10m were changed. Describe the transient and accident analyses to which these changes were affected and explain the need for these changes.

NMC Response: The response to Attachment 3 RAI #42 addresses the TS 3.4 bases change.

The analyses generating the safety limit curves and defining the DNBR parameter limits are related to TS 2.1 (safety limit curves) and TS 3.10.k, 3.10.l, and 3.10.m (DNBR parameter limits). These TS have been moved to the COLR (Reference 1) which will be in place for Cycle 26 operation. The changes are made to maintain consistency between the safety analysis assumptions and methods and the TS (COLR) requirements.

The safety limit curves show the loci of points of thermal power, reactor coolant system pressure, and reactor coolant system average temperature for which either the DNBR is equal to the DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. The safety limit curves are based on the nuclear hot channel factor limits of TS 3.10.b (these limits have also been moved to the COLR, see Reference 1).

The limits for the parameters that affect DNBR, core average temperature (TS 3.10.k), reactor coolant system pressure (TS 3.10.l), and reactor coolant flow (TS 3.10.m), are consistent with the safety analysis. By adhering to these operating parameter limits the plant is maintained within the safety analysis envelope. This assures that the DNBR limits are satisfied should a postulated design basis accident occur.

The bases for the safety limit curves (TS 2.1) and the DNBR parameter limits (TS 3.10.k, 3.10.l, and 3.10.m) remain the same. The values for these limits, however, are changed (see LAR 187 Attachment 9, Sections 2.1, 2.2, and 2.11) to be consistent with the revised safety analysis of LAR 187, the WRB-1 DNBR correlation, and the WRB-1 correlation limit. The DNBR-related transients of the safety analyses are analyzed consistent with these changes. The DNBR-related transients are described in attachment B.

References:

1. NMC Letter to NRC, from Mark E. Warner to Document Control Desk, entitled "License Amendment Request 185 To The Kewaunee Nuclear Power Plant Technical Specifications, "Core Operating Limits Report Implementation", Letter NRC-02-064, dated July 26, 2002 (TAC No. MB5717).

41. TS Basis 2.1-1 indicates that the settings of the power operated relief valves (PORVs), have been established to prevent the primary system exceeding its safe limit of 2735 psig for all transients except the RCCA ejection accident. Identify the transient in which the PORVs are credited for mitigating the overpressure event. Please confirm that the PORVs and their associated controls at KNPP are designed to safety grade standard so that the PORVs are qualified for accident mitigation.

NMC Response: The pressurizer PORV's and their associated controls are not designed to safety grade requirements and qualified for primary system pressure control and pressure transient mitigation. Therefore, the pressurizer PORV's are not credited in any of the design basis safety analyses for accident mitigation of overpressure events. The statement in the TS basis explains the function of the PORV's to understand their role in the defense-in depth-scheme if they were to operate on an overpressure condition even though they are not safety grade.

42. Please explain the need and the basis for changes made in TS B 3.4-2 regarding auxiliary feedwater system.

NMC Response: The proposed changes made in TS B3.4-2 regarding auxiliary feedwater system are being retracted from LAR 187 since the loss of normal feedwater (LONF) accident analysis, which was the basis for the TS B3.4-2 changes, is also being retracted from LAR 187 (see Attachment D). LONF safety analyses and any corresponding TS bases changes, if required, will be included in the Stretch Power Uprate License Amendment Request.

43. Please explain the basis for the increasing percentage of fuel rod experiencing DNB from 40 percent to 50 percent for a locked rotor event. Confirm that the radiological consequences are within the guidelines of 10 CFR 100 limits with such large amount of failed fuel.

NMC Response: Alternate Source Term (AST) Analyses (Reference 1) support the safety analyses of LAR 187. The AST Radiological Analysis for the Locked Rotor event conservatively modeled 100% fuel failure. The results of the AST Locked Rotor Radiological Analysis were acceptable (i.e. satisfied dose limits) at the 100% fuel failure assumption. Therefore it is acceptable to increase the allowed number of fuel rods experiencing DNB (these fuel rods are assumed to fail when they reach DNB) percentage from 40% to 50% since this percentage of fuel rods is bounded by the Locked Rotor Radiological Analyses input assumption for failed fuel rods.

With AST Radiological Analysis methods, the dose limits for KNPP safety analyses are changed from 10 CFR 100 limits to the fraction 10 CFR 50.67 limits specified in RG 1.183 (see Reference 1).

References:

1. Letter No. NRC-02-024, from Mark E. Warner to Document Control Desk, "Kewaunee Nuclear Power Plant, Revision to the Design Basis Radiological Analysis Accident Source Term," March 19, 2002.

44. Page 14-0-6 of the revised UFSAR indicated that there are changes to reactor protection setpoints (e.g. overpower delta T, over-temperature delta T, RCS low flow, pressurizer high level, and pressurizer low pressure). Please explain why these changes are not included in the list of proposed TS changes.

NMC Response: The only technical specification reactor trip set point values that are being changed for LAR 187 are the Overtemperature ΔT and the Overpower ΔT setpoints (see response to Attachment 3 RAI #2).

The changes to reactor protection setpoints shown on page 14.0.6 of the revised USAR (Attachment 5 of LAR 187) are changes to the safety analysis input assumption for reactor protection setpoints. The safety analysis reactor trip setpoint values bound the technical specifications setpoint values to provide adequate analytical margin. Since the revised safety analysis input values bound the current TS values and provide appropriate analytical margin, a change to the TS setpoint values is not required.

45. Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator (SG) water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15, 2002, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG water level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, ATWS and steam line break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affect the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how KNPP account for these uncertainties documented in these advisory letters in determining the SG water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the LOCA and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, the current analyses are still adequate.

NMC Response: The uncertainties in the steam generator (SG) water level setpoints and initial conditions associated with the phenomena discussed in Nuclear Safety Advisory Letters NSAL 02-3, NSAL 02-4, and NSAL 02-5 have been conservatively accounted for in the setpoint methodology and the in safety analysis methodology.

The SG water level process measurement accuracy (PMA) analysis explicitly accounts for the uncertainties due to the phenomena discussed in the NSALs noted above, as well as all other appropriate phenomena. The steam generator water level uncertainties resulting from the PMA are combined with the instrument channel uncertainties from the instrument uncertainty analysis to determine conservative values for the reactor protection system and engineered safety features system plant setpoints related to SG water level.

The plant setpoints for the reactor protection system and engineered safety features system related to SG water level are conservatively bounded by the corresponding safety analysis setpoints. In addition, the phenomena discussed in the NSALs mentioned above have been considered in determining the initial SG water level for all of the LOCA and non-LOCA safety analyses, including ATWS. Note that for analyses for which the acceptance criteria have been demonstrated to be insensitive to moderate changes in initial SG water level, nominal values for initial SG water level are used for both statistical and non-statistical analyses. For analyses for which acceptance criteria may be impacted by changes in initial SG water level, the uncertainties are applied in the conservative direction (e.g., LONF, MSLB, etc.).

The specific implementation of the setpoint methodology and the safety analysis methodology for KNPP demonstrates that all SG water level setpoint values (plant settings, technical specification values, and safety analysis values) are conservative and have sufficient margin (analytical, technical specification, and operating) to accommodate the SG water level uncertainties discussed in the NSALs mentioned above. In addition, the specific implementation of the safety analysis methodology demonstrates that the initial SG water levels used in the safety analyses conservatively bound the uncertainties discussed in the NSALs mentioned above and are therefore adequate.

46. Please explain why the following events are not included in the transient and accident analyses: 1) feedwater line break accident, 2) steam generator tube rupture accident, and 3) inadvertent startup of emergency core cooling system at power.

NMC Response: The feedwater line break and inadvertent startup of emergency core cooling system at power accidents are not design basis accidents for KNPP and therefore are not included in the transient and accident analyses. The Steam Generator Tube Rupture Safety Analysis supporting LAR 187 is included in Reference 1. The steam generator tube rupture accident will be reanalyzed for power uprate and will be presented in the Stretch Power Uprate License Amendment Request.

References:

1. Letter No. NRC-02-024, from Mark E. Warner to Document Control Desk, "Kewaunee Nuclear Power Plant, Revision to the Design Basis Radiological Analysis Accident Source Term," March 19, 2002.

47. Provide evaluation of the impacts of the proposed fuel changes and future power uprate on the ability of KNPP to cope with a Station Blackout event.

NMC Response: The proposed fuel change does not impact the station blackout (SBO) event analysis. The event is driven by decay heat and the analysis of the event uses the approved decay heat model based on 1979 ANS standard decay heat curve plus two sigma uncertainty (References 1, 2 and 3). The decay heat model is generically applicable to PWR fuel and is not dependent on the specific fuel design. The decay heat model is therefore applicable to the Westinghouse 422V+ fuel design, the Framatome ANP fuel design and any transition core. Therefore, the existing station blackout analysis remains applicable and bounding for KNPP during the transition to and operation with Westinghouse 422V+ fuel.

The measurement uncertainty recapture (MUR) power uprate of 1.4% will not impact the station blackout event analysis. An uprate could potentially impact the ability of the plant to withstand and recover from a station blackout due to the increased decay heat that must be removed from the RCS. However, the methodology and assumptions associated with the existing SBO analysis afford an exchange between the allowed power level and uncertainty. The methodology and assumptions involving equipment operability are not impacted by the MUR uprate. For example, there is no change in the ability of the turbine-driven auxiliary feedwater pump (supplied with steam from the steam generators) to support reactor heat removal due to the MUR uprate. The existing SBO analysis (References 1, 2 and 3) is performed with an operating power level of 1650 MWt plus 2% uncertainty, or 1683 MWt. This power level bounds the MUR uprate power level of 1673 MWt plus 0.6% uncertainty. Therefore, the existing station blackout analysis remains applicable and bounding for KNPP for the MUR uprate.

Additionally, the Technical Specification (TS) minimum required volume in the condensate storage tanks (CST) of 39,000 gallons remains acceptable for the MUR power uprate. The assumed power level and uncertainty and the TS CST volume are described in an NRC safety evaluation dated November 20, 1990 (Reference 1) and confirmed in a supplemental safety evaluation and an additional safety evaluation dated October 1, 1991 (Reference 2) and November 19, 1992 (Reference 3), respectively.

The stretch power uprate to 7.4% will impact the station blackout event analysis so the SBO event will be re-analyzed for the stretch uprate. Submittal of the stretch power uprate LAR is expected in March of 2003. The only expected change that the stretch uprate SBO analysis will require is that the Technical Specification minimum required volume in the CST to be increased from the current 39,000 gallons to 41,500 gallons.

References:

1. Letter to Ken H. Evers from NRC, "Safety Evaluation of the Kewaunee Nuclear Power Plant Response to the Station Blackout Rule (TAC No, 68558)," November 20, 1990.
2. Letter to Ken H. Evers from NRC, "Supplemental Safety Evaluation of the Kewaunee Nuclear Power Plant, Response to the Station Blackout Rule (TAC No. 68558)" October 1, 1991.
3. Letter to C.A. Schrock from NRC, "Kewaunee Nuclear Power Plant, unit No. 1 – Station Blackout Rule (10 CFR 50.63) (TAC No. M84521)," November 19, 1992.

48. To show that the referenced generically approved Best Estimate LOCA analysis methodologies apply specifically to the Kewaunee plant provide a statement that NMC and its vendor have ongoing processes which assure that LOCA the values and ranges of analysis inputs for peak cladding temperature- sensitive parameters bound the values and ranges of the as-operated plant for those parameters.

NMC Response: NMC and Westinghouse have ongoing processes which assure that the values and ranges of the LOCA analysis inputs for peak cladding temperature-sensitive parameters bound the values and ranges of the as-operated plant for those parameters.

49. If Kewaunee is sharing BE large-break loss-of-coolant accident (LBLOCA) analyses with any other plant(s), are the Kewaunee "plant-specific" analyses based on the model and or analyses of any other plant? If, so justify that the model or analyses apply to Kewaunee. (E.g. if the other design has a different vessel internals design the model wouldn't apply to Kewaunee.)

NMC Response: Kewaunee is not sharing its BELOCA analysis with another plant. The modeling used in the BELOCA analysis is specific to the Kewaunee RCS layout, ECCS configuration, and reactor vessel/internals.

1. References: Letter No. NRC-02-024, from Mark E. Warner to Document Control Desk, "Kewaunee Nuclear Power Plant, Revision to the Design Basis Radiological Analysis Accident Source Term," March 19, 2002.

50. Is NMC applying the same Kewaunee "plant-specific" analyses to any of the other units? If so please justify the applicability of the Kewaunee analyses to those plants.

NMC Response: No, the Kewaunee BELOCA analysis is not being applied to any other units.

51. For the loss of external electrical load transient at 102 percent power, your revised UFSAR indicates that you use 2200 psia for the initial condition when modeling the pressurizer pressure. Given that you operate at 2250 psia and that your uncertainty is ±50psi, explain how the 2200 psia assumption is more limiting than a 2300 psia for the overpressure condition. Also, if you use 2300 psia as the starting condition, what peak pressure is achieved? Is this pressure below 110% of your design values?

NMC Response: A negative uncertainty is applied to the initial pressurizer pressure because it conservatively delays the time of reactor trip due to high pressurizer pressure, i.e., it maximizes the duration of the primary-to-secondary power mismatch. Therefore, the last two questions are not applicable.

52. Discuss the basis for assuming an initial pressurizer pressure of 2250 for safety analyses concerning DNB. Why is this assumption resulting a most conservative minimum departure from nucleate boiling (MDNB) value?

NMC Response: The basis is that the RTDP analysis method was employed. With this method, pressure, temperature, flow and power uncertainties are statistically combined with the DNBR correlation uncertainties and uncertainties associated with nuclear and thermal parameters, fuel fabrication parameters, and codes to define the design limit DNBR (see WCAP-11397-P-A).

53. Discuss the basis for assuming a reactor vessel coolant flow of 186,000 gpm as the most conservative conditions in calculating MDNB value.

NMC Response: The basis is that the RTDP analysis method was employed. With this method, pressure, temperature, flow and power uncertainties are statistically combined with the DNBR correlation uncertainties and uncertainties associated with nuclear and thermal parameters, fuel fabrication parameters, and codes to define the design limit DNBR (see WCAP-11397-P-A).

54. With respect to the proposed uprated power level, please demonstrate that your LOCA analysis models and your emergency procedures are consistent with 10 CFR 50.46 requirements (1) during the switch-over from the refueling water storage tank to the containment emergency sump and (2) during long term cooling while the emergency core cooling system (ECCS) pumps draw water from the containment emergency sump. Note the general intention of this request is to obtain confirmation that ECCS pump operation, flow rates, and water injection locations will be consistent with 10 CFR 50.46 requirements.

NMC Response: Long-term core cooling requires adequate ECCS flow to provide core cooling. The confirmation of adequate ECCS flow during the cold leg recirculation period is based on the following assumptions:

- The current Westinghouse SBLOCA analysis methodology explicitly models ECCS flow interruptions and/or enthalpy changes during the switchover from injection phase to sump recirculation.

- The Westinghouse long-term core cooling methodology assumes that LB ECCS flow is not adversely affected by the switchover from injection phase to sump recirculation. For LBLOCA, adversely affected might mean the loss of all or a significant portion of the low head ECCS flow. In cases where ECCS flow is adversely affected (for example plant designs where the low head pumps supply both ECCS recirculation and containment spray flow), evaluations are performed to confirm that there is adequate core cooling for the LBLOCA scenario. These evaluations use minimum ECCS flow criteria that provide the removal of the heat generated in the core for the earliest possible entrance into sump recirculation.

  - For cold leg breaks, the large volume of LHSI flow to the upper plenum will make up core boil-off and metal heat and will establish reverse flow through the core.

  - Similarly, for hot leg breaks, the large volume of LHSI flow to the upper plenum will make up core boil-off, metal heat and entrainment out the break and natural circulation will preclude boron buildup.

  When calculating heat generated in the core to establish ECCS flow criteria, the decay heat generation rate is based on 1.2 times the 1971 ANS Standard for an infinite operating time consistent with Section 1.A.4 of 10CFR50 Appendix K. The decay heat generation includes a core power multiplier to address instrumentation uncertainty as identified by Section I.A of Appendix K.

55. With respect to the uprated power generation rate, please describe the containment emergency sump configuration, including the screen design, and show that an adequate water source is provided to the ECCS during operation after initiation of recirculation from the emergency sump. Your response should include consideration of the effect of potential debris accumulation on the effective water pressure and flow rate available to the ECCS pump suctions.

NMC Response: This RAI is related to power uprate only and will be addressed in the Stretch Power Uprate LAR.

56. Has the NRC approved the Westinghouse Best Estimate LBLOCA methodology for analyses of Framatome fuel in any other 2-loop upper plenum injection plant?

NMC Response: No. The Reference 1 3- and 4-loop BELOCA methodology has previously been accepted by the staff for the Catawba and McGuire units, References 2 and 3, respectively. The BELOCA analysis performed for the Catawba and McGuire units also addressed a transition from Framatome ANP fuel to Westinghouse fuel, as described in Reference 4. There is nothing unique to the Reference 5 2-loop BE UPI methodology that would invalidate the Catawba and McGuire precedent. The BE methodology for the 2-loop UPI plants is simply an extension of the methodology that is applied to 3- and 4-loop cold leg ECCS injection units to address the thermal-hydraulic phenomena unique to the UPI plant design.

The analytical process to account for mixed fuel types is discussed in response to Attachment 3 RAI No. 57.

References:

1. S. M. Bajorek, et al., WCAP-12945-P-A (Proprietary), Westinghouse Code Qualification Document for Best-Estimate Loss-of-Coolant Accident Analysis, Volume I, Rev. 2, and Volumes II-V, Rev. 1, and WCAP-14747 (Non-Proprietary), March 1998.
2. C. P. Patel (NRC) to G. R. Peterson (DEC), Catawba Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MA8719 AND MA8720), October 2, 2000.
3. F. Rinaldi (NRC) to H. B. Barron (DEC), McGuire Nuclear Station, Units 1 and 2 RE: Issuance of Amendments (TAC NOS. MA8719 AND MA8720), September 22, 2000.
4. M. S. Tuckman (DEC), Catawba Nuclear Station, Units 1 and 2 and McGuire Nuclear Station, Units 1 and 2 Implementation of Best-Estimate Large Break LOCA Methodology, April 10, 2000.
5. S. I. Dederer, et al., WCAP-14449-P-A, Application of Best-Estimate Large-Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Rev. 1 (Proprietary) and WCAP-14450-NP-A, Rev. 1 (Non-Proprietary), October 1999.

57. Discuss how the Kewaunee LBLOCA methodology(ies) account for the mixed fuel types.

NMC Response: The Framatome fuel was explicitly analyzed with WCOBRA/TRAC for two transition core scenarios. One option modeled the hot rod/assembly as fresh Westinghouse 422V+ surrounded by partially depleted Framatome fuel. The second option modeled the hot rod/assembly as partially depleted Framatome fuel surrounded by Westinghouse 422V+ fuel. For both options, the fuel in the low power peripheral region was modeled as partially depleted Framatome fuel. The calculated peak cladding temperature (PCT) for both cases was less than that calculated for a full core of Westinghouse 422V+ fuel, thus, there is no PCT penalty to assess for the mixed core transition cycles.

A detailed comparison of the mechanical design differences between the two types of fuel assemblies is shown in Table 2.1 of Attachment 4 of the Submittal. These differences have been explicitly modeled in the LBLOCA analysis. For SBLOCA, the mechanical differences are not important since hydraulic mismatch is not a factor for the transient. With this, the SBLOCA was analyzed using a full 422V+ core.

58. Because it is possible for one fuel type to be PCT-limiting, and another to be oxidation-limiting, provide a commitment to report LOCA analysis results for all fuel types (represented in a significant number of assemblies), per 10 CFR 50.46 (a)(3).

NMC Response: The BELOCA transition core analysis for Kewaunee showed that the WCOBRA/TRAC calculated PCT for the partially depleted Framatome ANP fuel was non-limiting by 125F at the limiting second reflood peak. This trend carries throughout the portion of the transient when significant oxidation would occur. In the SBLOCA analysis, it was determined that the PCT calculated for a full core of Westinghouse 422V+ fuel bounds the transition cycles. Given the low SBLOCA PCT of 1030F, cladding oxidation is not a concern.

The Framatome ANP fuel type was analyzed as though it would operate at the same peaking factors as the Westinghouse fuel (FQ = 2.50 and FH = 1.800). However, the Framatome ANP fuel will continue to have Tech Spec peaking factor limits of FQ = 2.35 and FH = 1.700. The combination of lower PCT and lower peaking factors makes it clear that the Framatome ANP fuel could never lead the core in either PCT or oxidation.

Without crediting the lower Tech Spec peaking factors for the Framatome ANP fuel, the 95th percentile reflood PCT for the Framatome ANP fuel is estimated to be 1959F. Consistent with the current 50.46 reporting practice, only the PCT for the full core analysis is reported. Had it been determined that there was a transition core PCT penalty, then the PCT assessment would have been added to the full-core analysis results and reported during the transition cycles. Since the LOCA analyses for a full-core of 422V+ fuel bound the transition cycles, no commitment will be made to report LOCA analysis results for non-limiting fuel types.

Figure 58-1 shows the PCT transients for a full core of 422V+ fuel compared to a transition core with partially depleted Framatome ANP fuel surrounded by 422V+. This calculation was performed using the reference transient conditions as defined by the Westinghouse methodology in WCAP-14449-P-A. Figures 58-2 and 58-3 compare the RCS pressure transients for those two runs, for the entire transient, and for the reflood portion, respectively. Figure 58-4 compares the hot assembly collapsed liquid level. It can be seen that the core configuration has little effect on the RCS pressure or hot assembly collapsed liquid level.

The reflood PCTs from Figure 58-1 are 1763°F for the full core 422V+ calculation, and 1638°F for the calculation with partially depleted Framatome ANP fuel surrounded by 422V+. This difference of 125°F was calculated assuming that the Framatome ANP fuel would operate at the same peaking factors as the Westinghouse fuel (F_Q = 2.50 and $F_{\Delta H}$ = 1.80). However, the Framatome ANP fuel will continue to have Tech Spec peaking factor limits of F_Q = 2.35 and $F_{\Delta H}$ = 1.70. Based on power distribution studies performed for the Westinghouse fuel, it is estimated that the lower peaking factors would reduce PCT for the Framatome ANP fuel by roughly 130°F.

The uncertainty methodology described in WCAP-14449-P-A was applied to the full core configuration, resulting in an estimated 95th percentile PCT of 2084°F. Oxidation was assessed using a transient more severe than the 95th percentile, resulting in a maximum local oxidation of 8.4%. As noted above, the estimated PCT for the Framatome ANP fuel in a transition core is approximately 255°F (125 + 130 = 255) lower than for the full core configuration. Oxidation is a strong function of cladding temperature. A detailed assessment of the oxidation for the Framatome ANP fuel is not required, as it is clearly bounded by the detailed assessment performed for a full core of Westinghouse.

The SBLOCA analysis was analyzed assuming a full core with fresh 422V+ fuel assemblies. This is bounding for transition cycles with partially depleted Framatome/ANP fuel. The PCT is 1030°F with a local cladding oxidation of 0.01% for the limiting case (3-inch break initiated at the High-Tavg value of 583.0°F).

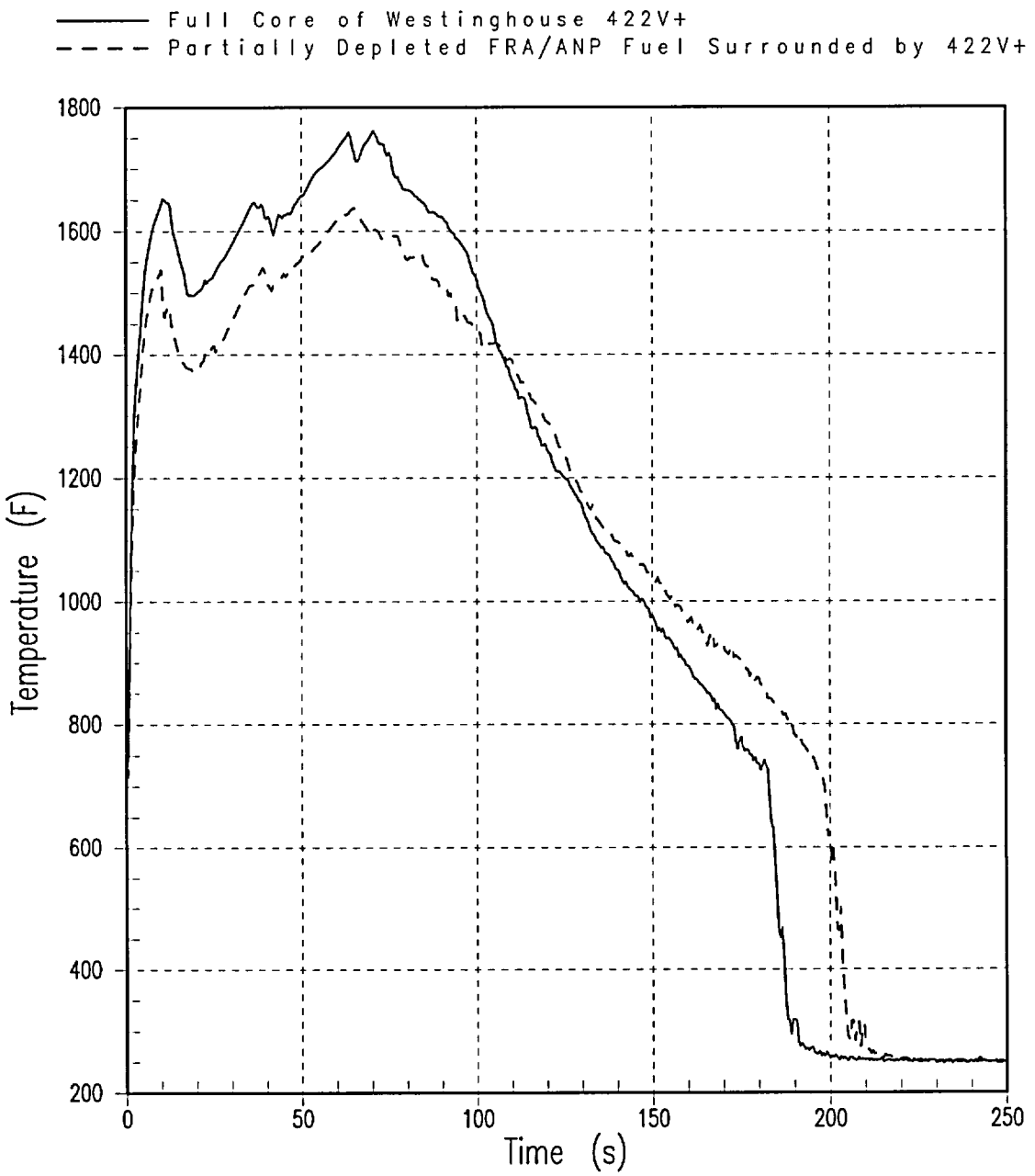


Figure 58-1: Peak Cladding Temperature Comparison

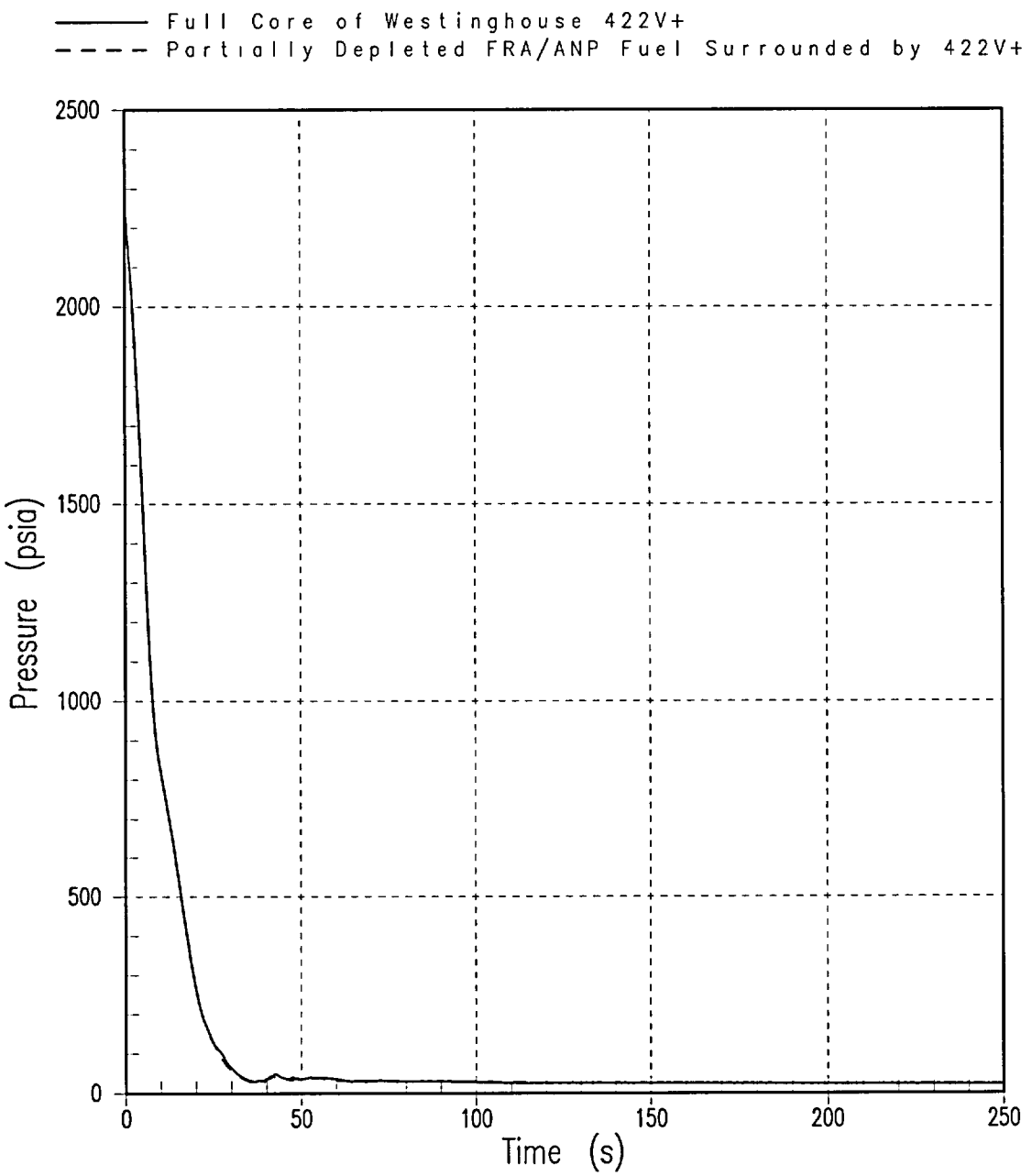


Figure 58-2: Full Transient RCS Pressure Comparison

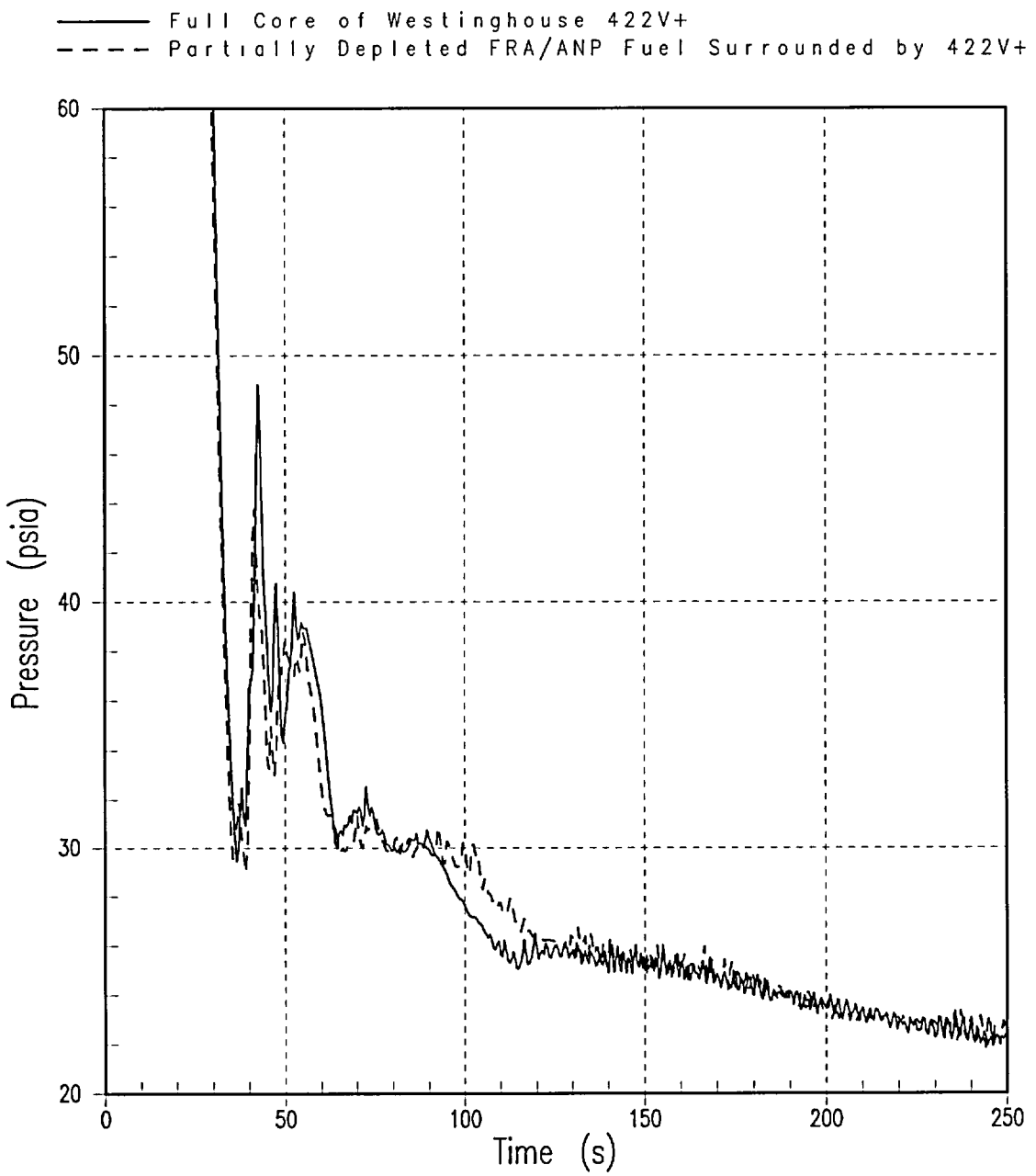


Figure 58-3: Reflood RCS Pressure Comparison

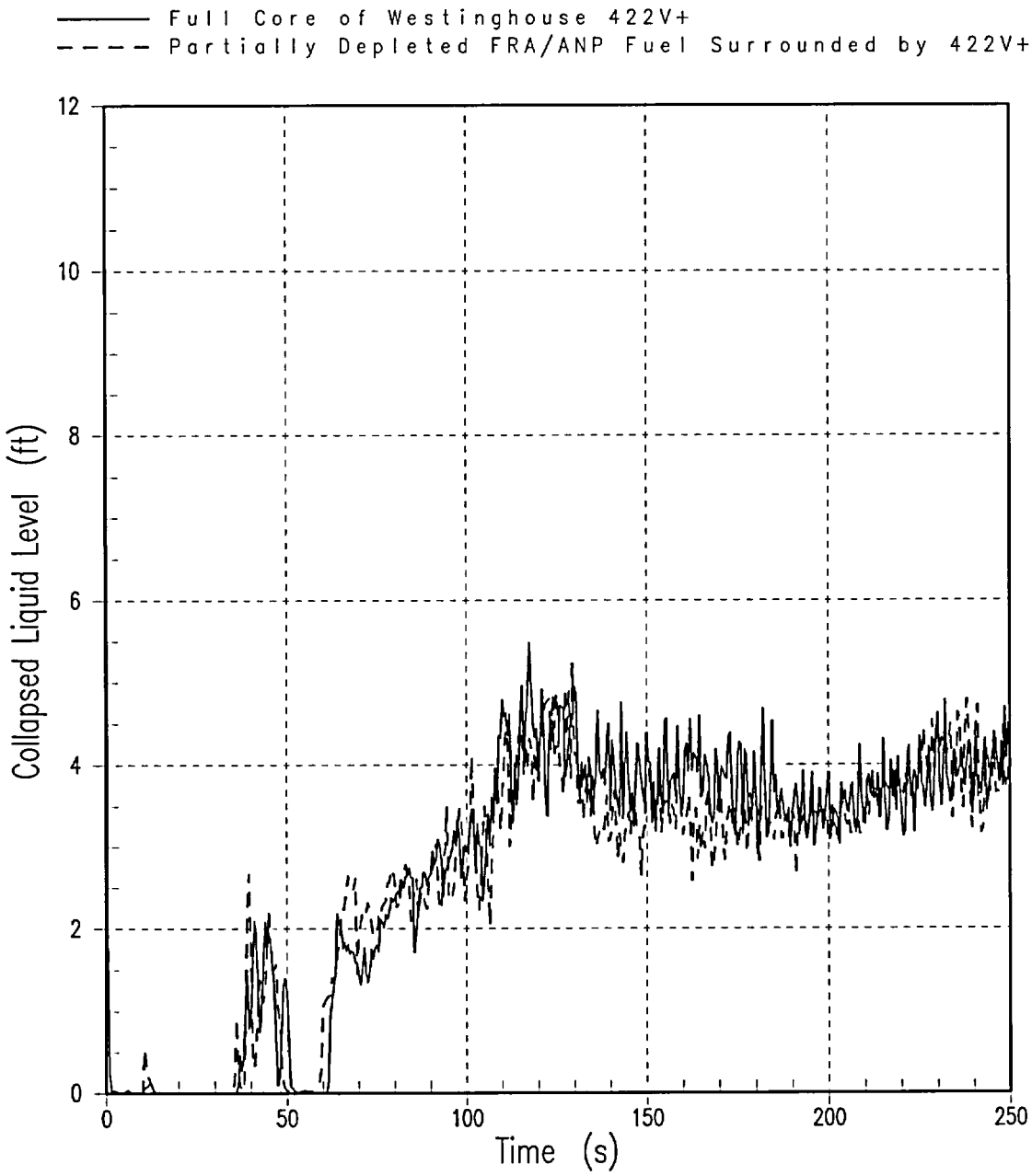


Figure 58-4: Hot Assembly Liquid Level Comparison

WCAP-15591, Rev 0, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology – Kewaunee Nuclear Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturis and 54F Replacement Steam Generator."
Questions

1. Provide a plant-specific basis for the parameter uncertainties and sensitivities listed in Tables 1 through 9 of WCAP-15591, including the source of each value and/or the method by which each value was determined.

NMC Response: The uncertainties associated with the Kewaunee instrumentation are based on manufacturer's specifications. Calibration tolerances for the Kewaunee instrumentation are based on Kewaunee calibration procedures. The magnitude of an instrument drift allowance is equivalent to its plant procedure calibration tolerance.

2. Kewaunee Assessment Process (KAP) No. 00-2991: Update updated safety analysis report (USAR) to Reflect Actual reactor coolant system (RCS) Bypass Flow Loop Indication and Alarm [UCR #R16-046 - Page 7.2-34] (ADAMS Accession No. ML003775364) stated, "This KAP was initiated to evaluate the discrepancy between the USAR and the actual reactor coolant system (RCS) bypass flow loop indication and alarms. The USAR [Updated Safety Analysis Report] described a common low flow alarm with individual status lights for each RCS bypass loop. The actual configuration has individual loop alarms and computer inputs, with no status lights. The function of the alarms/indication, to provide immediate indication of a low flow condition in the bypass loops, was met by the current plant configuration. . . ."

   In the discussion on page 9 of WCAP-15591, Westinghouse stated, "Allowances are made (as noted on Table 2) for hot leg and cold leg streaming, resistance temperature detector (RTDs), turbine pressure transmitter, process racks/indicators and controller. Based on one T_H and one T_C RTD per channel to calculate T_{avg} and with the RTDs located in the hot and cold leg bypass manifolds, the calculated electronics uncertainty using Equation 3 is [ ]$^{+a,c}$."

   Provide a technical basis for including hot leg temperature streaming in the uncertainty calculation.

NMC Response: The Kewaunee plant uses RTD bypass loops for hot leg and cold leg water temperature measurements. Three (3) scoops are located approximately 120 degrees apart around the hot leg pipe to collect the hot leg water and the temperature is measured by RTDs in the hot leg manifold. One (1) scoop collects the cold leg water and the temperature is measured by RTDs in the cold leg manifold. The hot leg water temperature streaming uncertainties account for variations in the hot leg water temperature collected by the hot leg scoops located around the circumference of the large hot leg pipes.

3. In the discussion on page 9 of WCAP-15591, Westinghouse stated, "A bias of [ ]$^{+a,c}$ for T_{cold} streaming (in terms of T_{avg}) based on a conservative [ ]$^{+a,c}$ T_{cold} streaming uncertainty is included in Table 2. . . ."

   Provide the technical basis for the cold leg temperature streaming bias, including a discussion of cold leg temperature streaming effects in KNPP.

NMC Response: A cold leg water temperature variation (referred to as temperature streaming) was discovered at a Westinghouse-designed plant in 1991. As a result, Westinghouse notified all operators of Westinghouse-designed plants of this effect. NMC was notified by Westinghouse in letter WPS-91-142. Based on this letter, a 2 deg-F cold leg temperature penalty (or a 1 deg-F Tavg temperature penalty) is included in the Tavg uncertainty reported in WCAP-15591. The effect is an additional 1 deg-F penalty on Tavg that is included in the Kewaunee Updated Safety Analysis Report.

4. On page 14 of WCAP-15591, Westinghouse states, "The thermal output of each steam generator is determined by a secondary side calorimetric measurement that is defined as: . . . . The steam enthalpy is based on the measurement of steam generator outlet steam pressure assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and a calculated feedwater pressure." A similar statement regarding the calculation of feedwater pressure was made on page 26 in the discussion describing the calculation of reactor power.

   On page 15, Westinghouse states, "The calorimetric RCS flow measurement is thus based on the following plant measurements: . . . Feedwater pressure (P_f) . . . ." Footnote (2) in Table 3 states that feedwater pressure is measured with a transmitter (PT-21196) and digital acquisition system on the feedwater bypass loop.

   Explain why a calculated feedwater pressure value is used in the secondary side calorimetric calculation, and a measured feedwater pressure value is used in the calorimetric RCS flow measurement uncertainty calculation and in the calculation of reactor power.

NMC RESPONSE: A high-accuracy flow measurement section is installed in a feedwater bypass loop at Kewaunee and it is capable of measuring full power feedwater flow. The feedwater bypass loop is used only during the calorimetric RCS flow measurement that is performed between 90 and 100% power and after each refueling. The feedwater bypass loop must be manually valved in and out, and its operation is controlled by plant procedure. A feedwater flow differential pressure measurement, a feedwater pressure measurement, and a feedwater temperature measurement are installed on the feedwater bypass loop. The feedwater pressure measurement is used only for a density correction to the feedwater flow differential pressure measurement. For the secondary side calorimetric power measurement portion of the calorimetric RCS flow measurement, the measured loop steam pressure is used as the loop feedwater pressure.

5. Provide a derivation of the equation for the controller deadband variance $(s_2)^2$ that is provided on page 9 of WCAP-15591.

NMC Response: The derivation of the equation for the controller deadband variance can be found in "Probability, Statistics, and Data Analysis", p. 104, 1971 by O. Kempthorne.

6. On page 16 of WCAP-15591, Westinghouse states, "For the measurement of feedwater flow, the feedwater venturi is calibrated by the vender in a hydraulics laboratory under controlled conditions to an accuracy of [ ]$^{+a,c}$. The calibration data that substantiates this accuracy is provided to the plant by the vendor. An additional uncertainty factor of [ ]$^{+a,c}$ is included for installation effects, resulting in a conservative overall flow coefficient (K) uncertainty of [ ]$^{+a,c}$ . . . ." A similar discussion is presented on page 26.

Provide a discussion of the basis for the additional uncertainty value that accounts for installation effects.

NMC RESPONSE: Since the accuracy of the venturi is determined in a laboratory, it is standard Westinghouse practice to double the uncertainty to account for installation effects such as sufficient straight length pipe upstream of the venturi.

7. On page 16 of WCAP-15591, Westinghouse states, "The uncertainty applied to the feedwater venturi thermal expansion correction (F_a) is based on the uncertainties of the measured feedwater temperature and the coefficient of thermal expansion for the venturi material, usually 304 stainless steel. For this material, a change of ±1 F in the nominal feedwater temperature range changes F_a by [ ]$^{+a,c}$ and the steam generator thermal output by the same amount.

"Based on data introduced into the ASME Code, the uncertainty in F_a for 304 stainless steel is ±5%. This results in an additional uncertainty of [ ]$^{+a,c}$ in feedwater flow. Westinghouse uses the conservative value of [ ]$^{+a,c}$." A similar discussion is presented on page 27.

Describe the process by which the ASME Code uncertainty in F_a for 304 stainless steel is translated into the stated additional uncertainty in feedwater flow.

NMC Response: Various references, such as the ASME Fluid Meters handbook, define a flow area thermal expansion term (F_a) of about 1.0070 for a 304 stainless steel flow meter at 450°F, a feedwater temperature that is higher than used at most (W)-designed PWRs. The uncertainty of ±5% on F_a applies to (α) in the thermal expansion term: $[1 + \alpha T]$. Therefore, the uncertainty would be 5% of αT, or 0.05 * 0.0070 = 0.00035, which is 0.035% of the nominal thermal expansion. For all RTDP analyses, the uncertainty was first rounded up to 0.04% and then provided with an arbitrary margin for possible subsequent uncertainty changes, resulting in 0.06%.

8. On page 16 of WCAP-15591, Westinghouse states, "Feedwater venturi ΔP uncertainties are converted to percent feedwater flow using the following conversion factor:

$$\text{percent flow} = (\Delta P \text{ uncertainty})(1/2)(\text{transmitter span}/90)^2\text{"}$$

On page 23 of WCAP-15591, Westinghouse states, ". . .The ΔP transmitter uncertainties are converted to percent flow using the following conversion factor:

$$\text{percent flow} = (\Delta P \text{ uncertainty})(1/2)(\text{Flow}_{max}/\text{Flow}_{nominal})^2$$

where Flow_{max} is the maximum value of the loop RCS flow channel."

On page 27 of WCAP-15591, Westinghouse states, "Feedwater venturi ΔP uncertainties are converted to percent flow using the following conversion factor:

$$\text{percent flow} = (\Delta P \text{ uncertainty})(1/2)(Flow_{max}/Flow_{nominal})^2$$

The feedwater flow transmitter span ($Flow_{max}$) is 117.5% of nominal flow."

Provide a derivation of the conversion equation, and provide a basis for each term in the parentheses.

NMC RESPONSE: % flow = (delta-P uncertainty)(1/2)($Flow_{max}/Flow_{nominal}$)^2 is the standard industry equation to convert a delta-P measurement uncertainty to a % flow measurement uncertainty. Since flow and delta-P are not a linear relationship, the conversion factor depends on the maximum value of the instrument range ($Flow_{max}$) and the nominal flow ($Flow_{nominal}$).

9. On page 17 of WCAP-15591, Westinghouse states, "The uncertainty on system heat losses, which is essentially all due to charging and letdown flows, has been estimated to be [ ]$^{+a,c}$ of the calculated value. Since direct measurements are not possible, the uncertainty on component conduction and convection losses has been assumed to be [ ]$^{+a,c}$ of the calculated value. Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics test performed at Prairie Island Unit 2 and by input power measurements from several other plants. Therefore, the uncertainty for the pump heat addition is estimated to be [ ]$^{+a,c}$ of the best estimate value. Considering these parameters as one quantity, which is designated the net pump heat addition uncertainty, the combined uncertainties are less than [ ]$^{+a,c}$ of the total, which is [ ]$^{+a,c}$ of core power."

   Provide a technical basis for each of the estimated uncertainties, and the process by which these uncertainties were combined and then subsequently used to determine the uncertainty value for core power.

NMC Response: The uncertainty for each term is calculated, and these uncertainties are combined to define the total net heat addition uncertainty. For the RTDP analyses, the net heat addition uncertainty is arbitrarily increased to ±20%. The table below summarizes net heat addition parameters for a 2-loop plant similar to the Kewaunee Nuclear Plant.

Summary of Net Heat Losses and Uncertainties			
Component	Heat Gain/Loss	Uncertainty	Uncertainty
RCPs	+30.215 MBtu/hr	±5%	±1.511 MBtu/hr
Charging/letdown	-4.391 MBtu/hr	±10%	±0.439 MBtu/hr
Other systems	-0.634 MBtu/hr	±10%	±0.063 MBtu/hr
Components	-0.883 MBtu/hr	±50%	±0.442 MBtu/hr
Total	+24.307 MBtu/hr	±10%	±2.455 Mbtu/hr
Total	+7.1 MW	±20%	±1.4 MW

The calculated net heat addition of 7.1 MW is applicable to the Kewaunee Nuclear Plant and would be applied when defining reactor core power during operation. The uncertainty of ±1.4 MWT is ±0.08% of the 1757 MWT NSSS power, and has a negligible impact on the power measurement uncertainty of ±1.72% or the flow measurement uncertainty of ±2.67% stated in WCAP-15591. Note that the bounding net heat addition of 8 MWT stated in WCAP-15591 is used where appropriate in RTDP analyses.

10. On page 17 of WCAP-15591, Westinghouse states, ". . . The uncertainties for the instrumentation are noted in Table 3 and the sensitivities are provided on Table 4. The hot leg streaming is split into random and systematic components. For Kewaunee where the RTDs are located in RTD bypass manifolds, the hot leg temperature streaming uncertainty components are [ ]$^{+a,c}$ random and [ ]$^{+a,c}$ systematic."

    Provide the basis for the hot leg temperature streaming uncertainty component values.

NMC Response: Hot leg temperature streaming uncertainties were based on an evaluation of hot leg circumferential temperature measurements at a 2-loop plant and two 3-loop plants. Subsequent measurements at a 4-loop plant, obtained prior to implementing core low leakage loading patterns (LLLPs), supported the conclusions obtained from this evaluation. Additional measurements evaluated after implementing LLLPs showed that measured hot leg temperatures were biased high (conservative), thus reducing or eliminating the non-conservative impact of hot leg streaming uncertainties.

A hot leg streaming uncertainty exists when the temperature gradient across the pipe is non-linear, since measurement of a linear gradient with three temperatures (either hot leg scoops or thermowell RTDs) results in no measurement uncertainty. Therefore, the objective of the evaluation was to define an uncertainty based on a conservative non-linear gradient. Several hot leg temperature gradient patterns were defined, using the maximum measured circumferential temperature gradient and its maximum slope as a guide in defining non-linear gradients. The difference between the gradient pattern average temperature and the average of the three temperatures defined the streaming uncertainty for a case. The evaluation considered changes in uncertainty as a gradient pattern was rotated within the hot leg pipe, thereby covering all possible gradient orientations for a specific pattern. For the RTD bypass system, the evaluation also considered uncertainties caused by imbalanced flows in the three scoop bypass lines.

Since measured loop gradients and loop bypass line configurations tended to be similar at a plant, part of the uncertainty for a loop was considered to be systematic. Small differences in loop gradients and bypass lines were assumed to result in a random uncertainty for a loop. The systematic uncertainty was based on the maximum average of uncertainties defined by gradient rotation, and on an allowance for systematic differences in bypass line flow resistances. The calculated systematic uncertainty was then increased by 50% to obtain the current value of ±0.65°F. The random uncertainty was based on the maximum variation defined by gradient rotation. The calculated random uncertainty was also increased by 50% to obtain the current value of ±0.65%.

11. On page 17 of WCAP-15591, Westinghouse used the values from Table 5 in the 2-loop uncertainty equation (with biases) to calculate the flow uncertainty. The Table 5 bias values appear to be inconsistent with the resulting stated flow uncertainty bias value.

    Provide the calculation that results in the flow uncertainty bias value provided on page 18.

NMC Response: To determine the total bias on page 18, refer to the bias components on page 22.

12. On page 23 of WCAP-15591, Westinghouse states, ". . . The loop RCS flow uncertainty is then combined with the calorimetric RCS flow measurement uncertainty. This combination of uncertainties results in the following total flow uncertainty:

# of loops	flow uncertainty (% flow)
2	±2.86 (random)
	±0.11 (bias)

The corresponding value used in RTDP is:

# of loops	standard deviation (% flow)
2	[ ]$^{+a,c}$

A similar discussion regarding feedwater venturi ΔP uncertainties is presented on page 27.

Provide the calculation details that resulted in the percent flow uncertainties and the percent flow standard deviation.

NMC Response: The random flow uncertainty on page 23 is a square-root-sum-of-the-squares of the random uncertainty components on page 24. The bias on page 23 is the bias from the calorimetric RCS flow measurement on page 22. The percent flow standard deviation for a 95% probability value at a 95% confidence level is equal to the total random uncertainty divided by 1.96.

ATTACHMENT B

NUCLEAR MANAGEMENT COMPANY, LLC
KEWAUNEE NUCLEAR PLANT
DOCKET 50-305

February 27, 2003

Letter from Thomas Coutu (NMC)

To

Document Control Desk (NRC)

License Amendment Request 187a

5 ACCIDENT ANALYSIS

The transient safety analyses discussed herein support the fuel transition to the 0.422-inch outer diameter (OD) VANTAGE + 14x14 fuel assembly with PERFORMANCE + features, hereinafter referred to as 422V+ fuel. In addition, the non-loss-of-coolant-accident (LOCA) safety analyses presented in Section 5.1 also support the 7.4-percent power uprating program, with the exception of the anticipated transients without scram (ATWS). The fuel transition and power uprating design modifications are justified with respect to the non-LOCA and LOCA design bases in Sections 5.1 and 5.2, respectively. Section 5.3 contains the containment design basis analysis and Section 5.4 addresses radiological analysis. The systems and components analyses/evaluations are covered in Section 6.0. Key analysis assumptions or bases that differ from those specified below are identified as required.

Key fuel features applicable to all analyses that have been considered include:

- Gadolinia (2, 4, 6, or 8 weight percent [w/o]) fuel burnable absorber
- Pre-oxide coated lower fuel rods (with the exception of the Gadolinia rods)
- ZIRLO™ fuel cladding, ZIRLO™ instrumentation and guide thimble tubes, and ZIRLO™ mid-grids
- Mechanical dimensional design changes for high burnup
- 0.422-inch OD fuel rod design
- Low cobalt top and bottom nozzles
- Solid, mid-enriched pellets in axial blankets (Gadolinia rods are annular)
- Cold, undensified fuel stack height of 143.25 inches
- New optimized-fuel-assembly (OFA) style low-pressure drop (LPD) mid-grid design for the 0.422-inch OD fuel rod

5.1 NON-LOCA TRANSIENTS

This section summarizes the non-LOCA transient analyses and evaluations performed to support the implementation of the 422V+ fuel transition and power uprate programs at the Kewaunee Nuclear Power Plant (KNPP).

5.1.0.1 Fuel Design Mechanical Features

The effects of fuel design mechanical features on the non-LOCA transient analyses are accounted for in fuel-related input assumptions, such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

As demonstrated in Sections 2, 3, and 4, the Westinghouse VANTAGE + 14x14 fuel with PERFORMANCE + features (422V+) and the Framatome HTP Heavy fuel (FRA-ANP) currently in use at KNPP are of very similar design and are mechanically, thermal-hydraulically, and neutronically compatible. Also, the non-LOCA transient analyses Nuclear Steam Supply System (NSSS) models utilize only a limited amount of detail in the fuel-related input assumptions (see above). In addition, the non-fuel-related acceptance criteria parameter results of the non-LOCA analyses (for example, RCS pressure, MSS pressure, pressurizer does not become water solid, etc.) are not overly sensitive to the fuel-related input assumptions. Therefore, the results of the non-LOCA transient analyses that are not fuel-related are applicable to transition cores as well as full 422V+ cores. Furthermore, the thermal-hydraulic statepoints (RCS pressure, core inlet temperature, core flow, and core-average heat flux) generated in the non-LOCA transient analyses are suitable for use in the thermal-hydraulic MDNBR analyses of FRA-ANP fuel.

5.1.0.2 Peaking Factors, Kinetics Parameters

The power distribution is characterized by an enthalpy hot channel factor (radial peaking, $F^N_{\Delta H}$) of 1.64 (Revised Thermal Design Procedure [RTDP (Reference 5-1)])/1.70 (non-RTDP) and a heat flux hot channel factor (total peaking, F_Q) of 2.50 for the 422V+ fuel. $F^N_{\Delta H}$ is important for transients that are departure from nucleate boiling (DNB) limited (Note that Table 5.1-1 identifies those events analyzed for DNB concerns as well as the DNB methodology used: (RTDP or non-RTDP)). Since $F^N_{\Delta H}$ increases with decreasing power level (due to rod insertion), all transients that may be DNB limited are assumed to begin with an $F^N_{\Delta H}$ consistent with the $F^N_{\Delta H}$ defined in the Technical Specifications for the nominal power level. F_Q is important for transients that may be overpower limited. F_Q may increase with decreasing power level such that the full power hot-spot heat flux is not exceeded (that is, F_Q x power equals the design hot-spot heat flux). Consequently, all non-LOCA transients for this Fuel Upgrade/Power Uprating (FU/PU) Program that may be overpower limited assume an initial hot full power F_Q of 2.50.

The analyses of events that are sensitive to minimum shutdown margin assume 1.30-percent $\Delta k/k$ for end-of-cycle (EOC) conditions (steam line break) and 1.0-percent $\Delta k/k$ for beginning-of-cycle (BOC) conditions (chemical and control volume system malfunction). The fact that the shutdown margin (SDM) varies as a function of core burnup is consistent with the current KNPP Technical Specifications.

5.1.0.3 Uprating Program Features

The uprating program features that were considered include:

- A nuclear steam supply system (NSSS) power level of 1780 MWt (~7.4-percent uprate) (including 8 MWt of pump heat)
- A reactor coolant vessel average temperature (T_{avg}) range from 556.3°F to 573.0°F
- A reactor coolant system (RCS) thermal design flow (TDF) of 178,000 gpm
- A steam generator tube plugging (SGTP) level of up to 10 percent; a maximum loop-to-loop tube plugging asymmetry of 10 percent has been addressed.
- A nominal operating RCS (pressurizer) pressure of 2250 psia

For most transients that are DNB limited, the initial conditions are assumed to be at nominal values. The uncertainty allowances for power, temperature, pressure, and flow are included statistically in the departure from nucleate boiling ratio (DNBR) limit value by using the RTDP (Reference 5-1) methodology. Transient analyses in which the RTDP is employed apply an RCS flow that is consistent with the minimum measured flow (MMF) of 186,000 gpm. The MMF is the Technical Specifications minimum flow measurement requirement.

For transient analyses that are not DNB limited, or for which RTDP is not employed, the initial conditions are obtained by applying the maximum, steady-state errors to the nominal values in the most conservative direction (Standard Thermal Design Procedure [STDP]). These cases all assume an RCS flow consistent with the TDF.

The following steady-state errors are considered in the non-LOCA transient analyses:

- The NSSS power allowance applied for calorimetric error is a conservative ± 2 percent.
- The T_{avg} allowance for dead band and system measurement errors is ± 6.0°F.
- The pressurizer pressure allowance for steady-state fluctuations and measurement errors is ± 50.1 psi.

5.1.0.4 Other Major Assumptions

Table 5.1-2 lists the non-LOCA initial condition assumptions used. Other major assumptions considered in the non-LOCA transient analyses are discussed below:

a. A ± 1-percent setpoint tolerance and a +3 percent accumulation have been considered in the modeling of the main steam safety valves (MSSVs). Staggered lift setpoints are modeled for the MSSVs (using plant-specific Technical Specification setpoints), as shown in Table 5.1-3.

b. The pressurizer safety valves (PSVs) are modeled assuming a ± 1-percent setpoint tolerance. Additionally, when it is conservative to do so (that is, for peak RCS pressure concerns), the

effects of the PSV loop seals are explicitly modeled, as discussed in Reference 5-2. See Table 5.1-3 for more information.

c. Consistent with the KNPP Technical Specifications, a + 5 pcm/°F moderator temperature coefficient (MTC) is assumed up to 60-percent power (Technical Specification 3.1.f). A zero MTC is modeled at power levels greater than 60 percent.

d. The fission product contribution to decay heat assumed in the non-LOCA analyses is the American Nuclear Society (ANS)-5.1-1979 residual decay heat model discussed in Reference 5-3, increased by two standard deviations for conservatism.

e. Since KNPP is not licensed to operate at-power with a single reactor coolant pump (RCP) in operation (Technical Specification 3.1.a.1), the non-LOCA analyses for the FU/PU Program only address two-loop operation.

f. The assumed core bypass flow percentages are 5.5 percent for RTDP analyses and 7.0 percent for STDP analyses.

5.1.0.5 Overtemperature- and Overpower-ΔT Reactor Trip Setpoints

The overtemperature and overpower ΔT (OTΔT/OPΔT) reactor trip setpoints were recalculated for the FU/PU Program assuming the most conservative core thermal limits based on using the RTDP, as described in Section 4.0 of this Reload Transition Safety Report (RTSR). These core limits are applicable to 422V+ fuel, assuming the uprated core power of 1772 MWt and a nominal RCS pressure of 2250 psia. The core thermal limits used to calculate the OTΔT/OPΔT setpoints are provided in Figure 5.1-1 and in the corresponding Technical Specification figure. These figures are applicable to the 422V+ fuel assemblies. The OTΔT and OPΔT trip setpoints calculated for the FU/PU Program are illustrated in Figure 5.1-2. These setpoints remain constant (with the exception of T', which depends on the operating T_{avg} chosen) and are applicable across the entire range of RCS average temperatures being considered for this program (that is, T_{avg} window between 556.3°F and 573.0°F). The revised safety analysis setpoint functions are also presented in Table 5.1-4.

The adequacy of these setpoints is confirmed in the analyses of those events that credit these functions for accident mitigation by showing that the DNB design basis is met. The revised safety analysis setpoints are based upon the assumption that the reference average temperatures (T') used in the OTΔT and OPΔT setpoint equations correspond to the selected operating temperature within the T_{avg} window. A temperature deviation of 2.6°F on T_{avg} was also included to account for the effects of hot leg and cold leg streaming on the indicated T_{avg}. This requirement is essential to ensure that the actual plant conditions required to generate an OTΔT and/or OPΔT trip signal are bounded by the assumptions made in the safety analyses.

The boundaries of operation defined by the OTΔT and OPΔT trips are represented as "protection lines" in Figure 5.1-2. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions, a trip would occur well within the area bounded by these lines. (These protection lines are based upon the safety analysis limit OTΔT and OPΔT setpoint values, which are essentially the Technical Specification nominal values with allowances for the adverse instrumentation

behavior, setpoint errors, and acceptable drift between instrument calibration.) The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the limit value (in this case, 1.34 for both typical and thimble cells). All points below and to the left of a DNB line for a given pressure have a DNBR greater than the safety analysis limit DNBR value. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable safety analysis limit DNBR at any point.

The area of permissible operation (power, temperature, and pressure) is bounded by the combination of reactor trips: high neutron flux (fixed setpoint); high and low RCS pressure (fixed setpoints); OTΔT and OPΔT (variable setpoints), and the opening of the MSSVs (modeled as a single valve with an opening setpoint of up to the highest MSSV setpoint plus 1-percent tolerance, 3-percent accumulation, and a 38 psi pressure drop between the steam generators and the MSSVs), which limit the maximum RCS average temperature. The DNBR safety analysis limit value, which was used as the limit value for all accidents analyzed using RTDP, is conservative compared to the actual DNBR design limit (1.24 and 1.24 for typical and thimble cells, respectively) required to meet the DNB design basis.

It should be noted that the revised OTΔT and OPΔT setpoint equations require a change to the temperature range for certain resistance temperature detector (RTD) instrumentation. The RTD ranges that are compatible with the revised setpoints are:

- $T_{cold} \Rightarrow 500 - 650°F$
- $T_{hot} \Rightarrow 500 - 650°F$
- $T_{avg} \Rightarrow 520 - 620°F$
- $\Delta T \Rightarrow 0 - 100°F$

It is recommended that the appropriate changes be made to the applicable setpoints document for the plant.

The changes associated with the OTΔT and OPΔT setpoints have been confirmed as being acceptable by showing that the DNB design basis is met for the DNB-limited transient analyses discussed herein.

5.1.0.6 RPS and ESFAS Functions Assumed in Analyses

Table 5.1-5 contains a list of the different reactor protection system (RPS) and engineered safety features actuation system (ESFAS) functions credited in the non-LOCA transient analyses. The safety analysis setpoints, as well as the time delays associated with each of these functions, are also presented in Table 5.1-5.

5.1.0.7 RCCA Reactivity Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod cluster control assemblies (RCCAs) and the variation in rod worth as a function of rod position. With respect to the non-LOCA transient analyses, the critical parameter is the time from beginning of RCCA insertion to dashpot entry, or approximately 85 percent of the RCCA travel, although negative reactivity addition

continues to be modeled until rods are completely inserted. For the non-LOCA analyses, the assumed insertion time from fully withdrawn to dashpot entry is the Technical Specification limit of 1.8 seconds.

Three figures relating to RCCA drop time and reactivity worth are presented in this report. The RCCA position (percent insertion) versus the time from release is presented in Figure 5.1-3. The normalized reactivity worth assumed in the safety analyses is shown in Figures 5.1-4 and 5.1-5. These present the worth versus rod insertion and time from release, respectively.

5.1.0.8 Reactivity Coefficients

The transient response of the RCS is dependent on reactivity feedback effects, in particular the MTC and the Doppler power coefficient (DPC). Depending upon event-specific characteristics, conservatism dictates the use of either maximum or minimum reactivity coefficient values. Justification for the use of the reactivity coefficient values is treated on an event-specific basis. Table 5.1-6 presents the core kinetics parameters and reactivity feedback coefficients assumed in the non-LOCA FU/PU analyses.

The maximum and minimum integrated DPCs assumed in the safety analyses are provided in Figure 5.1-6. Note that the steam line break core response analysis uses a different DPC based on an RCCA (not shown in Figure 5.1-6).

5.1.0.9 Computer Codes Utilized

Summary descriptions of the principal computer codes used in the non-LOCA transient analyses are provided below. Table 5.1-7 lists the computer codes used in each of the non-LOCA FU/PU analyses.

FACTRAN

FACTRAN (Reference 5-4) calculates the transient temperature distribution in a cross-section of a metal clad UO_2 fuel rod and the transient heat flux at the surface of the cladding, using as input the nuclear power and the time-dependent coolant parameters of pressure, flow, temperature, and density. The code uses a fuel model that simultaneously contains the following features:

- A sufficiently large number of radial space increments to handle fast transients such as a rod ejection accident
- Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation
- The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, Zircaloy-water reaction, and partial melting of the fuel

RETRAN

RETRAN (Reference 5-5) is used for studies of transient response of a pressurized water reactor (PWR) system to specified perturbations in process parameters. This code simulates a multi-loop system by a lumped parameter model containing the reactor vessel, hot- and cold-leg piping, reactor coolant pumps,

steam generators (tube and shell sides), main steam lines, and the pressurizer. The pressurizer heaters, spray, relief valves, and safety valves may also be modeled. RETRAN includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a detailed nodalization for the thermal transients. The RPS simulated in the code includes reactor trips on high neutron flux, OTΔT and OPΔT, low RCS flow, high and low pressurizer pressure, high pressurizer level, and lo-lo steam generator water level. Control systems are also simulated including rod control and pressurizer pressure control. Parts of the safety injection system (SIS), including the accumulators, may also be modeled. RETRAN approximates the transient value of DNBR based on input from the core thermal safety limits.

LOFTRAN

Transient response studies of a PWR to specified perturbations in process parameters use the LOFTRAN (Reference 5-6) computer code. This code simulates a multi-loop system by a model containing the reactor vessel, hot- and cold-leg piping, steam generators (tube and shell sides), the pressurizer and the pressurizer heaters, spray, relief valves, and safety valves. LOFTRAN also includes a point neutron kinetics model and reactivity effects of the moderator, fuel, boron, and rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients. The code simulates the RPS, which includes reactor trips on high neutron flux, OTΔT and OPΔT, high and low pressurizer pressure, low RCS flow, lo-lo steam generator water level, and high pressurizer level. Control systems are also simulated including rod control, steam dump, and pressurizer pressure control. The SIS, including the accumulators, is also modeled. LOFTRAN also approximates the transient value of DNBR based on the input from the core thermal safety limits.

ANC

ANC (Reference 5-7) is an advanced nodal code capable of two-dimensional and three-dimensional neutronics calculations. ANC is the reference model for certain safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worths, reactivity coefficients, etc. In addition, three-dimensional ANC validates one-dimensional and two-dimensional results and provides information about radial (x-y) peaking factors as a function of axial position. It can calculate discrete pin powers from nodal information as well.

TWINKLE

TWINKLE (Reference 5-8) is a multi-dimensional spatial neutron kinetics code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 8,000 spatial points and performs steady-state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. The code provides various outputs, such as channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures. It also predicts the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

VIPRE

The VIPRE computer program (Reference 5-9) performs thermal-hydraulic calculations. This code calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core.

5.1.0.10 Classification of Events

Each of the events listed in Table 5.1-8 is presented in Chapter 14 of the KNPP USAR. Each non-LOCA event is categorized with respect to its potential consequences. Those presented in Section 14.1 of the USAR are events that have no offsite-dose radiation consequences. These are typically events that may be expected to occur during a calendar year. The exception to these two criteria are the loss-of-flow events presented in Section 5.1.9 of this report (Section 14.1.8 of the USAR). This section includes the locked-rotor event which, based on its expected frequency of occurrence and potential radiological consequences, is considered a fault as serious as those presented in Section 14.2 of the USAR. However, because of the similarities between the analysis methodology used for this event and the partial and complete loss-of-reactor-coolant-flow events, which do satisfy both categorization criteria, all three cases are presented together, both in this report and in Section 14.1.8 of the USAR. The design requirements supported by the safety analyses presented in USAR Section 14.1 are as follows:

- Occurrences are accommodated with, at most, a reactor trip with the plant capable of returning to operation.
- Release of radioactive materials in effluent to unrestricted areas shall be in conformance with the Code of Federal Regulations (CFR) Section 10 CFR 20.
- These incidents shall not generate a more serious incident without other incidents occurring independently.
- There shall be no consequential loss of function of any barrier to the escape of radioactive products (no fuel rod failure or overpressurization).

The transients presented in USAR Section 14.2 and the locked rotor event presented in Section 14.1.8 address faults that are not expected to occur but are postulated because their consequences would include the potential for radioactive releases. The faults are the most drastic design-basis events. The design requirements supported by the safety analyses presented in USAR Section 14.2 are as follows:

- Release of radioactive material shall not result in any undue risk to public health and safety and does not exceed the guidelines of 10 CFR 100.
- There shall be no consequential loss of function of systems needed to cope with the event.

5.1.0.11 Events Evaluated or Analyzed

The effects of the FU/PU implementation on each of the USAR transients listed in Table 5.1-1 were evaluated or analyzed. These transient evaluations and analyses demonstrate that all applicable safety analysis acceptance criteria continue to be met for the intended FU/PU implementation at KNPP. Table 5.1-1 summarizes the results obtained for each of the non-LOCA transient analyses.

Markups of the KNPP Technical Specifications (Reference 5-10) and USAR (Reference 5-11) are provided in Appendices A and D of this report, respectively, to reflect incorporation of the FU/PU Program.

Table 5.1-1 Non-LOCA Analysis Limits and Analysis Results

USAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	(Limiting Case)
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Minimum DNBR below first mixing vane grid (non-RTDP, W-3 correlation)	1.39	1.588
		Maximum fuel centerline temperature, °F	4746[1]	2685
14.1.2	Uncontrolled RCCA Withdrawal at Power	Minimum DNBR (RTDP, WRB-1)	1.34	1.46
		Peak RCS pressure, psia	2750	< 2750
		Peak MS system pressure, psia	1210	1204
14.1.3	RCCA Misalignment	Minimum DNBR (RTDP, WRB-1)	1.34	> 1.34
14.1.4	Chemical and Volume Control System Malfunction			
	(at power)	Minimum time to loss of shutdown margin, minutes	15	22.68 (manual) 25.06 (auto)
	(during startup)	Minimum time to loss of shutdown margin, minutes	15	28.75
	(during refueling)	Minimum time to loss of shutdown margin, minutes	30	31.60
14.1.5	Startup of an Inactive Reactor Coolant Loop	Non-limiting event, no analysis performed		
14.1.6	Feedwater Temperature Reduction Incident	Analysis limit is feedwater (FW) ΔT for 10% load increase. Limiting case result is FW ΔT for the opening of the bypass valves that direct FW flow around the low-pressure FW heaters (NMC Scope).	73°F	33°F

Notes:

1. Melting temperature corresponding to 8-weight-percent Gadolinia fuel.

Table 5.1-1 (cont.) Non-LOCA Analysis Limits and Analysis Results

USAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	(Limiting Case)
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions	Minimum DNBR (RTDP, WRB-1)	1.34	1.709 (HFP) 2.837 (HZP)
14.1.7	Excessive Load Increase Incident	Minimum DNBR (RTDP, WRB-1)	1.34	> 1.34
14.1.8	Loss of Reactor Coolant Flow (PLOF/CLOF/UF)*	Minimum DNBR (RTDP, WRB-1)	1.34	1.646 (PLOF) 1.386 (CLOF) 1.420 (UF)
	Locked Rotor	Rods-in-DNB (RTDP, WRB-1)	See Section 5.4	
		Peak RCS pressure, psia	2750	2683
		Peak cladding temperature, °F	2700	1900
		Maximum Zirc-water reaction, %	16	0.61
14.1.9	Loss of External Electrical Load	Minimum DNBR (RTDP, WRB-1)	1.34	1.74
		Peak RCS pressure, psia	2750	2697
		Peak MS system pressure, psia	1210	1202
14.1.10	Loss of Normal Feedwater	*****Analysis Retracted*****		
14.1.11	ATWS	No analysis required to support the fuel transition (See Section 5.1.14)		
14.1.12	Loss of AC Power to the Plant Auxiliaries	Maximum pressurizer water volume ft^3	1010.1	698
14.2.5	Steam Line Break (Core response only)	Minimum DNBR (non-RTDP, W-3)	1.472	2.29

Note:
* PLOF = partial loss of flow
CLOF = complete loss of flow
UF = underfrequency

Table 5.1-1 Non-LOCA Analysis Limits and Analysis Results (cont.)

USAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	(Limiting Case)
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)			
	(BOC-HZP)	Maximum fuel pellet average enthalpy, cal/g	200	144.9
		Maximum fuel melt, %	10	0
	(BOC-HFP)	Maximum fuel pellet average enthalpy, cal/g	200	167.4
		Maximum fuel melt, %	10	2.17
	(EOC-HZP)	Maximum fuel pellet average enthalpy, cal/g	200	161.6
		Maximum fuel melt, %	10	0
	(EOC-HFP)	Maximum fuel pellet average enthalpy, cal/g	200	170.3
		Maximum fuel melt, %	10	5.89

Table 5.1-2 Non-LOCA Plant Initial Condition Assumptions for the KNPP Fuel Upgrade/Power Uprating Program

Nominal Conditions	RTDP	Non-RTDP	Notes
Core Power (MWt)	1772.0	1772.0 * 1.02	1
Nominal Total RCP Heat (MWt)	8.0	8.0	1,2
Maximum Full-Power Vessel T_{avg} (°F)	573.0	573.0 + 6.0	1,3
Minimum Full-Power Vessel T_{avg} (°F)	556.3	556.3 - 6.0	1,3
No-Load RCS Temperature (°F)	547.0	547.0	1,4
Pressurizer Pressure (psia)	2250	2250 ± 50.1	1
Steam Flow (lbm/hr)	see Note 5	see Note 5	5
Steam Pressure (psia)	see Note 5	see Note 5	5
Feedwater Temperature (°F)	437.1	437.1	1
Pressurizer Water Level (% span)	see Note 6	see Note 6	6
Steam Generator Water Level (% NRS)	44	44 ± 15	7

Notes:

1. Table 1-1 (PCWG-2707) Cases 1 through 4
2. Total RCP heat input minus RCS thermal losses
3. A full-power RCS T_{avg} window between 556.3°F and 573.0°F is supported. Some analyses only use one full-power T_{avg} value either because a clear direction of conservatism exists, or because the full-power T_{avg} does not have a significant effect on the analysis results.
4. All analyses assume a programmed no-load RCS T_{avg} of 547°F. For the events initiated from a no-load condition [rod withdrawal from subcritical, steam line break, rod ejection, boron dilution], the use of the no-load temperature as the initial temperature bounds the case of startup operations at KNPP being performed at a hot-zero power (HZP) temperature lower than 547°F. This is because the DNBR calculations and the boron dilution calculations would be less limiting at a lower RCS temperature.
5. The nominal steam flow rate and steam pressure depend on other nominal conditions See Table 1-2
6. The nominal pressurizer water level varies linearly from 21% of span at an RCS T_{avg} of 547°F to 34 4% of span at an RCS T_{avg} of 562 4°F for any RCS full-power T_{avg} less than or equal to 562.4°F. For an RCS full-power T_{avg} of 573°F, the nominal pressurizer water level varies linearly from 21% of span at an RCS T_{avg} of 547°F to 48% of span at the RCS T_{avg} of 573°F.
7. The nominal steam generator water level modeled in the analyses performed in support of the KNPP FU/PU Program is a constant 44% narrow range span (NRS), regardless of the power level

Table 5.1-3 Pressurizer and Main Steam System (MSS) Pressure Relief Assumptions			
		Pressure Relief Model Number*	
USAR Section	Event Description	Pressurizer	MSS
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	5	5
14.1.2	Uncontrolled RCCA Withdrawal at Power		
	DNB	1	3
	RCS Overpressure	2B	3
14.1.3	RCCA Misalignment	6	6
14.1.4	Chemical and Volume Control System Malfunction	5	5
14.1.5	Startup of an Inactive Reactor Coolant Loop	Analysis not required	
14.1.6	Feedwater Temperature Reduction Incident	5	5
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions	1	3
14.1.7	Excessive Load Increase Incident	5	5
14.1.8	Loss of Reactor Coolant Flow	2A	3
14.1.8	Locked Rotor	2B	3
14.1.9	Loss of External Electrical Load		
	DNB	1	3
	RCS Overpressure	2B	3
14.1.10	Loss of Normal Feedwater	**Analysis Retracted**	
14.1.11	Anticipated Transients Without Scram	Analysis not required	
14.1.12	Loss of AC Power to the Plant Auxiliaries	1	3
14.2.5	Steam Line Break	4	4
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	5	5

*The pressure relief models are described on the following pages.

Table 5.1-3 Pressurizer and Main Steam System Pressure Relief Assumptions (cont.)

Model 1 (maximum pressurizer pressure relief)

The pressurizer power-operated relief valve (PORV) setpoint is 100 psi above the initial pressure. The PORV relief rate is 179,000 lbm/hr per valve (2 valves total). The pressurizer spray valves begin to open when the indicated pressurizer pressure exceeds the initial value by 25 psi. The pressurizer spray valves are full open when the indicated pressurizer pressure exceeds the initial value by 75 psi. A linear increase in the pressurizer spray valve flow area is assumed between these points. The full-open spray valve flow area is 0.0376 ft^2.

The PSV setpoint is 1% below the nominal setpoint of 2485 psig. Once the PSVs come open, they do not reseat until the pressure drops 5% below the nominal setpoint. No time delay penalty is applied to account for the purging of the water in the PSV loop seals. The PSV relief rate is 345,000 lbm/hr per valve (2 valves total).

Model 2A (minimum pressurizer pressure relief)

The pressurizer PORVs and pressurizer sprays are assumed to be unavailable. The PSVs are modeled assuming the nominal setpoint of 2485 psig. However, the PSVs do not actuate.

Model 2B (minimum pressurizer pressure relief)

The pressurizer PORVs and pressurizer sprays are assumed to be unavailable.

The PSV setpoint is increased by 1% above the nominal set pressure of 2485 psig to account for set pressure tolerance. An additional 1% allowance is included to address the set pressure shift phenomena described in WCAP-12910 (Reference 5-2), *Pressurizer Safety Valve Set Pressure Shift*. A maximum time delay of 1.13 seconds is applied to account for the purging of the water in the PSV loop seals. The PSV relief rate is 345,000 lbm/hr per valve (2 valves total).

Table 5.1-3 Pressurizer and Main Steam System Pressure Relief Assumptions (cont.)

Model 3 (staggered MSSV setpoints)

There are 5 MSSVs on each loop with a total relief capacity of ~2128 lbm/sec (total of 10 valves) at 1181 psig. The assumed setpoints are listed below.

Valve Bank	Nominal Setpoint	Pressure at which MSSVs Open*
1	1074 psig	1151.96 psia
2	1090 psig	1168.60 psia
3	1105 psig	1184.20 psia
4	1120 psig	1199.80 psia
5	1127 psig	1207.08 psia

Note:
* Pressure includes +1% for setpoint tolerance, +3% for accumulation, +20 psi for pressure drop from the steam generator to the valve, and +15 psi to convert to atmospheric pressure

Note that the MSSV relief models that were used in the analyses for the loss of external electrical load and loss of normal feedwater/loss of AC power to the plant auxiliaries events vary slightly from that presented above. Refer to the accident-specific calculation notes for additional information.

Model 4

No specific pressurizer pressure or main steam system relief inputs are modeled. The pressurizer pressure and steam pressure both decrease during this event. Thus, the behavior of the pressurizer spray, relief valves, and safety valves, or the MSSVs, is irrelevant.

Model 5

Pressurizer and main steam system relief is not modeled because either the computer codes used for these analyses do not include pressurizer or steam generator models, or the analysis is a hand calculation that does not involve these plant components. Refer to accident-specific calculation notes for additional information.

Model 6

The generic (that is, not plant-specific) analysis performed to address this event assumes that the pressurizer PORVs actuate at 2350 psia with a total maximum relief capacity of 16.65 ft^3/sec. The pressurizer spray valve setpoints assumed are the same as those specified for Model 1, but the total spray capacity is 52.2 lbm/sec. The PSVs and MSSVs are modeled and assumed to be available, but do not actuate.

Table 5.1-4 KNPP Overtemperature and Overpower ΔT Setpoints	
Allowable T_{avg} Range	556.3°F to 573.0°F
K1 (safety analysis value)	1.30
K1 (Technical Specification value)	1.20
K2	0.0150
K3	0.00072
K4 (safety analysis value)	1.16
K4 (Technical Specification value)	1.095
K6	0.00103
T'	556.3°F to 573.0°F*
P'	2250 psia
f(ΔI) Deadband	-22% ΔI to +12% ΔI
f(ΔI) Negative Gain	0.86
f(ΔI) Positive Gain	0.96
High Pressurizer Pressure Reactor Trip Setpoint	2425 psia
Low Pressurizer Pressure Reactor Trip Setpoint	1850 psia

*Value to be set equal to less than or equal to the full power operating T_{avg} chosen

Table 5.1-5 Summary of RPS and ESFAS Functions Actuated

USAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical Condition	Power-range high neutron flux reactor trip (low setting)	35%	0.65
14.1.2	Uncontrolled RCCA Withdrawal at Power	Power-range high neutron flux reactor trip (high setting)	118%	0.65
		Overtemperature ΔT reactor trip	Table 5.1-4	Note 1
14.1.3	RCCA Drop	Low pressurizer pressure reactor trip	Note 2	2.0
14.1.4	Chemical and Volume Control System Malfunction	Overtemperature ΔT reactor trip	Table 5.1-4	Note 1
14.1.5	Startup of an Inactive Reactor Coolant Loop	NA	NA	NA
14.1.6	Feedwater Temperature Reduction Incident	NA	NA	NA
14.1.6	Excessive Heat Removal Due to Feedwater System Malfunctions (Feedwater Flow Increase)	Hi-Hi steam generator water level turbine trip with reactor trip on turbine trip	100% NRS	1.0 and 2.0, respectively

Table 5.1-5 Summary of RPS and ESFAS Functions Actuated (cont.)

USAR Section	Event Description	RPS or ESFAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
14.1.7	Excessive Load Increase Incident	NA	NA	NA
14.1.8	Loss of Reactor Coolant Flow	Low RCS loop flow reactor trip	86.5%	0.75
14.1.8	Locked Rotor	Low RCS loop flow reactor trip	86.5%	0.75
14.1.9	Loss of External Electrical Load	High pressurizer pressure reactor trip	2425 psia	1.0
		Overtemperature ΔT reactor trip	see Table 5.1-4 / Note 1	
14.1.10	Loss of Normal Feedwater	*****Analysis Retracted*****		
14.1.11	ATWS	NA	NA	NA
14.1.12	Loss of All AC Power to the Station Auxiliaries	Lo-Lo SG water level reactor trip	0% NRS	1.5
		Lo-Lo SG water level AFW pump start	0% NRS	60.0
14.2.5	Steam Line Break	Hi-Hi steam flow setpoint	~200% of nominal	NA
		Lo-Lo steam pressure safety injection (SI) setpoint	495 psia	0.5
		Steamline isolation delay		7.6
		Feedwater isolation delay		85.7
		SI pumps at full flow following SI signal		25.0 (w/ offsite power)
				30.0 (w/o offsite power)
14.2.6	Rupture of a Control Rod Mechanism Housing (RCCA Ejection)	Power-range high neutron flux reactor trip (low and high settings)	35% (low setting)	0.65
			118% (high setting)	0.65

Table 5.1-5 (cont.)	Summary of RPS and ESFAS Functions Actuated
Note 1:	The modeling of the OTΔT reactor trip includes a time constant (first order lag) of 4.0 seconds for the measurement of the vessel T_{avg} and ΔT. This lag accounts for the response of the RTDs, the RTD electronic filter (if any), the RTD bypass piping fluid transport delay, and the RTD bypass piping heatup thermal lag. In addition, a straight delay of 2.0 seconds is assumed which accounts for electronics delay, reactor trip breakers opening, and RCCA gripper release.
Note 2:	The generic two-loop dropped RCCA analysis, applicable to KNPP, models the low pressurizer pressure reactor trip setpoint as a convenience trip. The cases that actuate this function assume dropped rod and control bank worth combinations that are non-limiting with respect to DNB. The use of a plant-specific low pressurizer pressure setpoint that is lower than the value assumed in the generic analysis does not invalidate the generic two-loop statepoints evaluated as part of the KNPP FU/PU Program. Therefore, the low pressurizer pressure reactor trip setpoint value that is used in the generic two-loop dropped RCCA analysis (1860 psia) does not represent an analytical limit for this function for KNPP.

Table 5.1-6 Core Kinetics Parameters and Reactivity Feedback Coefficients

Parameter	BOL[1] (Min. Feedback)	EOL[2] (Max. Feedback)
Moderator Temperature Coefficient, pcm/°F	5.0 (below 60% RTP)	NA
	0.0 (above 60% RTP)	
Moderator Density Coefficient, Δk/(g/cc)	NA	0.50
Doppler Temperature Coefficient, pcm/°F	-0.91	-2.90
Doppler-Only Power Coefficient, pcm/%power (Q = power in %)	-12.0 + 0.045Q	-24.0 + 0.100Q
Delayed Neutron Fraction	0.0072 (maximum)	0.0043 (minimum)
Minimum Shutdown Margin, %Δk	1.0	1.30
Minimum Doppler Power Defect, pcm		
Rod Ejection	1000	900
Rod Withdrawal from Subcritical	1100	NA

Notes:
1. Beginning of life
2. End of life

Table 5.1-7 Summary of Initial Conditions and Computer Codes Used

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Reactor Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Uncontrolled RCCA Withdrawal from a Subcritical Condition	TWINKLE FACTRAN VIPRE	W-3[1] WRB-1[2]	No	0 (1772 MWt – core power)	79,922	547	2,160
Uncontrolled RCCA Withdrawal at Power	RETRAN	WRB-1	Yes (DNB) No (Pressure)	100 (DNB) 60 (DNB) 10 (DNB) 8 (Pressure) (1780 MWt – NSSS power)	186,000 (DNB) 178,000 (Pressure)	573.0 (100%) 562.6 (60%) 549.6 (10%) 555.6 (8%)	2,250 (DNB) 2,200 (Pressure)[3]
RCCA Misalignment	LOFTRAN (4) VIPRE	WRB-1	Yes	100	186,000	573.0	2,250
Chemical and Volume Control System Malfunction	NA	NA	NA	NA	NA	579.0 (power) 554.3 (startup) 140 (refueling)	2,250 (power) 2,250 (startup) 14.7 (refueling)
Startup of an Inactive Reactor Coolant Loop	KNPP Tech. Specs. prevent event occurrence						
Feedwater Temperature Reduction Incident	Bounded by excessive load increase						
Excessive Heat Removal Due to Feedwater System Malfunctions (Feedwater Flow Increase)	RETRAN VIPRE	WRB-1 (hot full power - HFP) W-3 (HZP)	Yes No	100 0 (1780 MWt - NSSS power)	186,000 178,000	573.0 547.0	2,250

Notes:
1. Below the first mixing vane grid
2. Above the first mixing vane grid
3. An additional 0 1 psi uncertainty has been evaluated
4. The LOFTRAN portion of the analysis is generic, the DNB evaluation performed with VIPRE utilizes the plant-specific values presented

Table 5.1-7 Summary of Initial Conditions and Computer Codes Used (cont.)

Accident	Computer Codes Used	DNB Correlation	RTDP	Initial Power, %	Reactor Coolant Flow, gpm	Vessel Average Coolant Temp, °F	RCS Pressure, psia
Excessive Load Increase Incident	NA	WRB-1	Yes	100 (1772 MWt - core power)	186,000	573.0	2,250
Loss of Reactor Coolant Flow	RETRAN VIPRE	WRB-1	Yes	100 (1780 MWt - NSSS power)	186,000	573.0	2,250
Locked Rotor	RETRAN VIPRE FACTRAN	WRB-1	No (Hot Spot) Yes (DNB)	102 (hot spot) 100 (DNB) (1780 MWt - NSSS power)	178,000 (hot spot) 186,000 (DNB)	579 0 (hot spot) 573.0 (DNB)	2,300 (hot spot)(1) 2,250 (DNB)
Loss of External Electrical Load	RETRAN	WRB-1	No (Overpressure) Yes (DNB)	102 (pressure) 100 (DNB) (1780 MWt - NSSS power)	178,000 (pressure) 186,000 (DNB)	579.0 (pressure) 573.0 (DNB)	2,200 (pressure)(1) 2,250 (DNB)
Loss of Normal Feedwater/Loss of AC Power to the Plant Auxiliaries	RETRAN	NA	No	102 (1780 MWt - NSSS power)	178,000	579.0	2,300(1)
ATWS	Analysis not required for fuel transition.						
Steam Line Break	RETRAN VIPRE	W-3	No	0 (1780 MWt - NSSS power)	178,000	547.0	2,250
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	TWINKLE FACTRAN	NA	No	102 (HFP) 0 (HZP) (1772 MWt - core power)	178,000 (HFP) 79,922 (HZP)	579.0 (HFP) 547.0 (HZP)	2,200(1)

Notes:

1. An additional 0.1 psi uncertainty has been evaluated

Table 5.1-8 Transients Evaluated or Analyzed for the KNPP Fuel Upgrade/Power Uprating Program			
Transient	**RTSR Section**	**USAR Section**	**Notes**
Uncontrolled RCCA Withdrawal from a Subcritical Condition	5.1.1	14.1.1	A
Uncontrolled RCCA Withdrawal at Power	5.1.2	14.1.2	A
RCCA Misalignment	5.1.3	14.1.3	E
Chemical and Volume Control System Malfunction	5.1.4	14.1.4	A
Startup of an Inactive Reactor Coolant Loop	5.1.5	14.1.5	E
Feedwater Temperature Reduction Incident	5.1.6	14.1.6	E
Excessive Heat Removal Due to Feedwater System Malfunctions	5.1.6	14.1.6	A
Excessive Load Increase Incident	5.1.7	14.1.7	E
Loss of Reactor Coolant Flow/Locked Rotor	5.1.8	14.1.8	A
Loss of External Electrical Load	5.1.9	14.1.9	A
Loss of Normal Feedwater	5.1.10	14.1.10	A
Loss of AC Power to the Plant Auxiliaries	5.1.11	14.1.12	A
Steam Line Break	5.1.12	14.2.5	A
Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)	5.1.13	14.2.6	A
Anticipated Transients Without Scram	5.1.14	14.1.11	E

Notes:
A = Complete Analysis
E = Evaluation

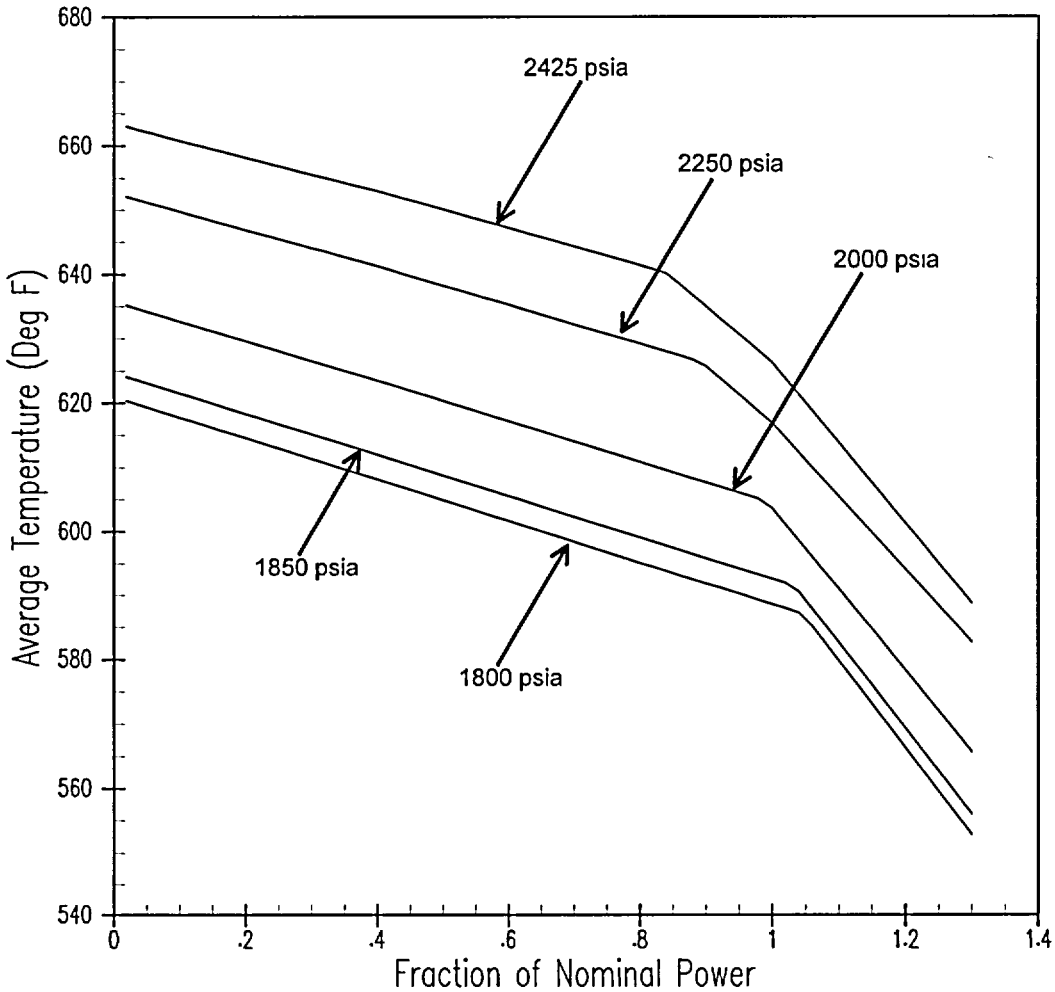


Figure 5.1-1 Reactor Core Safety Limit

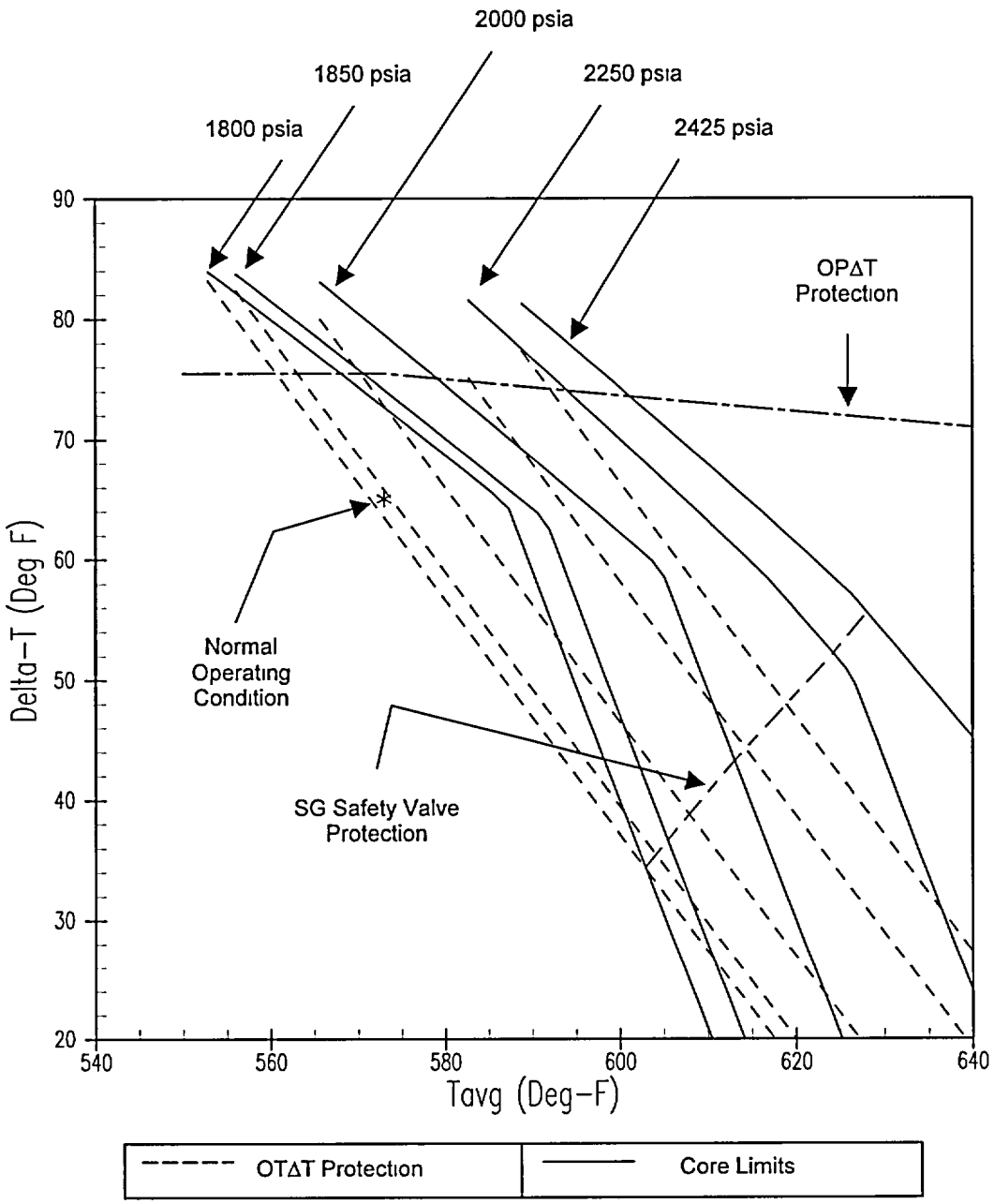


Figure 5.1-2 Illustration of Overtemperature and Overpower ΔT Protection

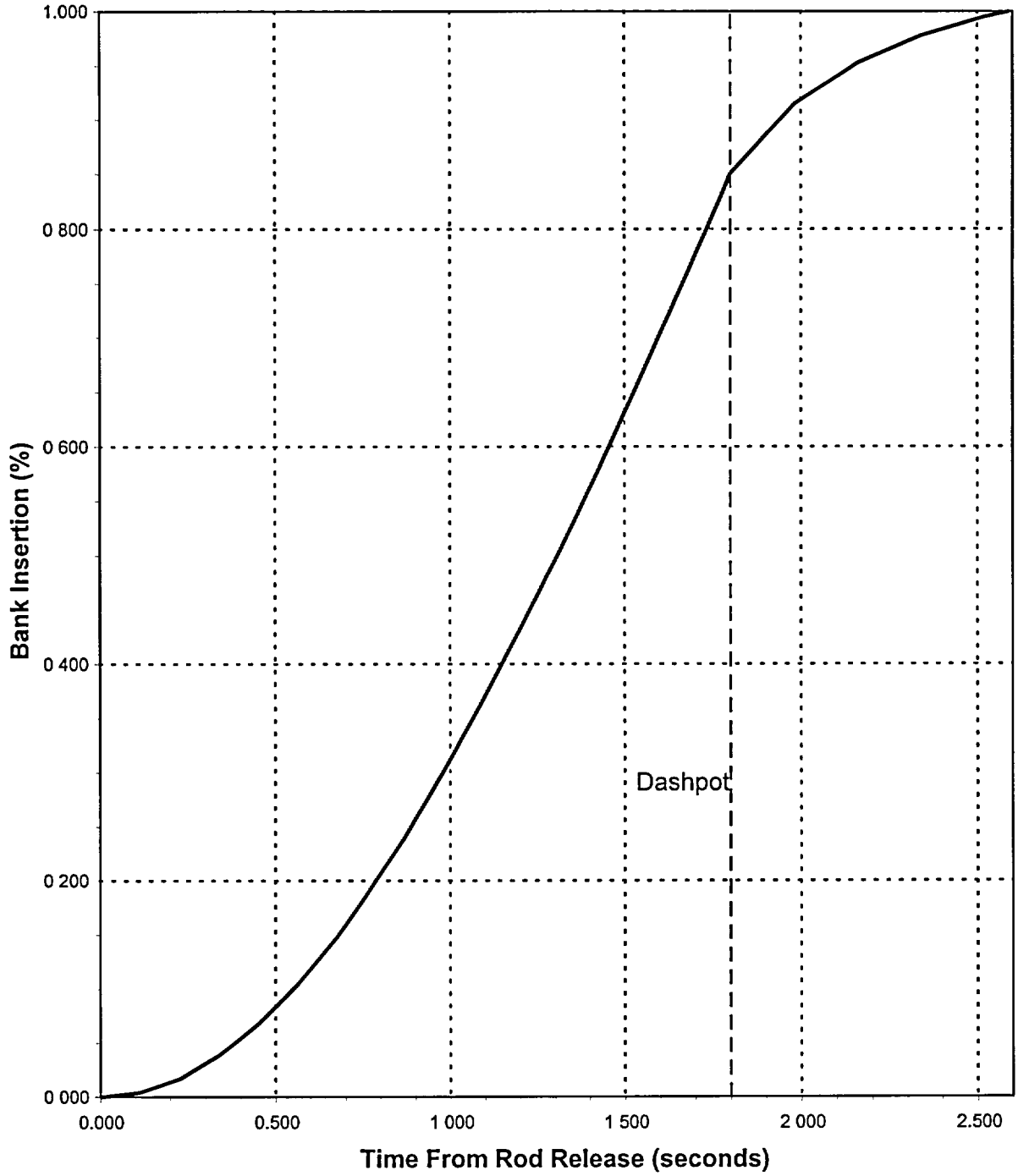


Figure 5.1-3 RCCA Position (Insertion) versus Time From Release

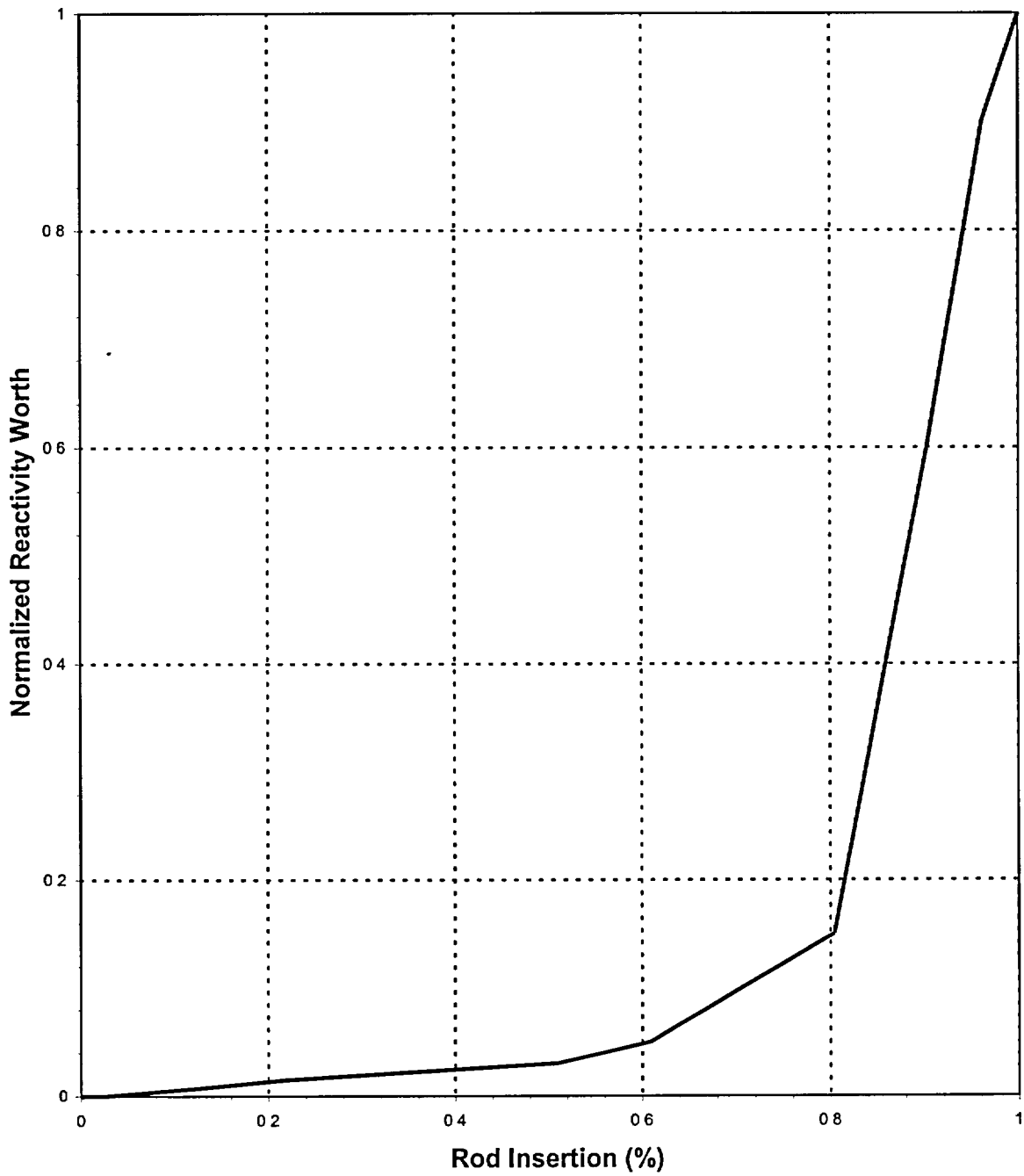


Figure 5.1-4 Normalized RCCA Reactivity Worth versus Percent Insertion

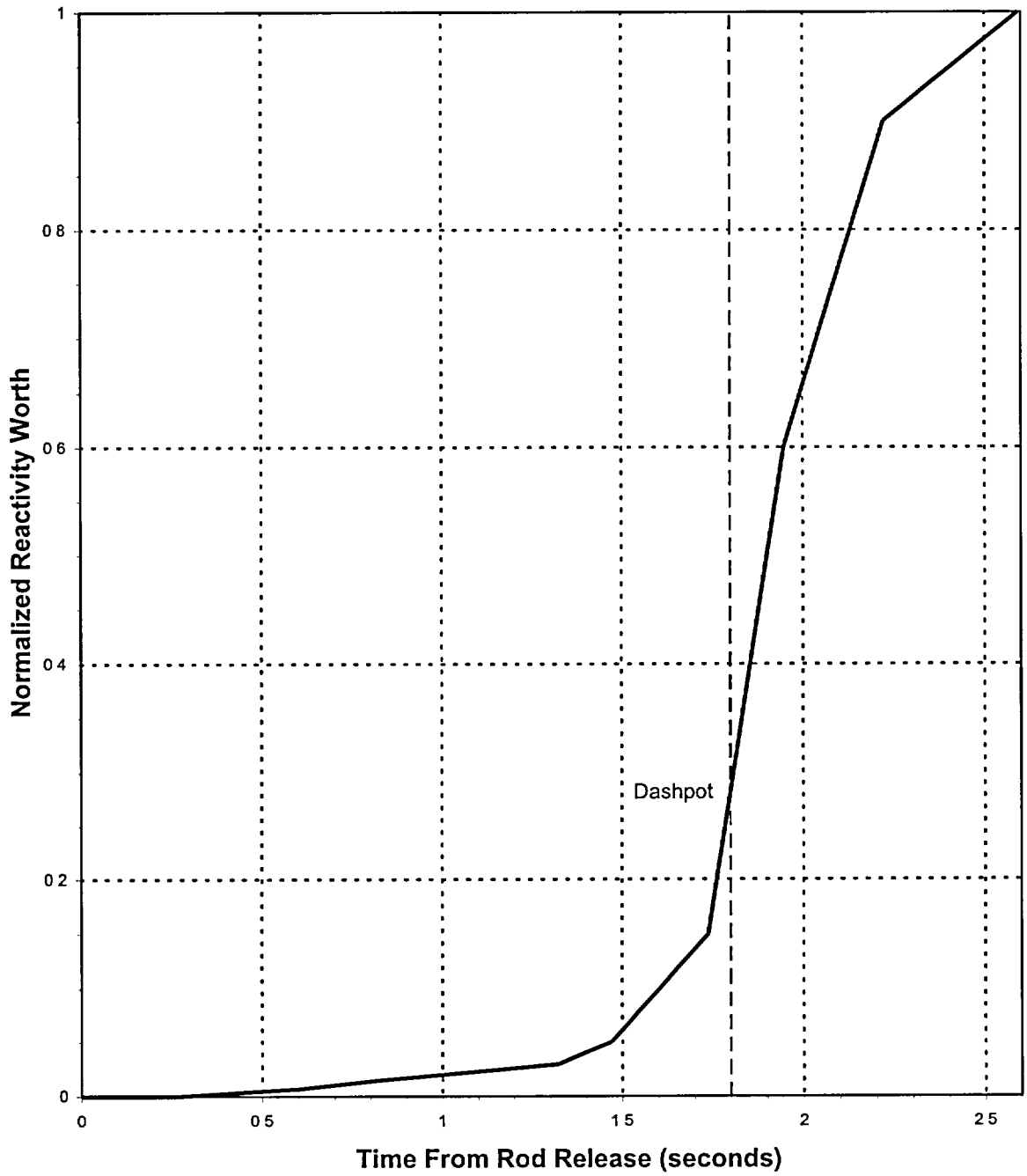


Figure 5.1-5 Normalized RCCA Bank Reactivity Worth versus Time

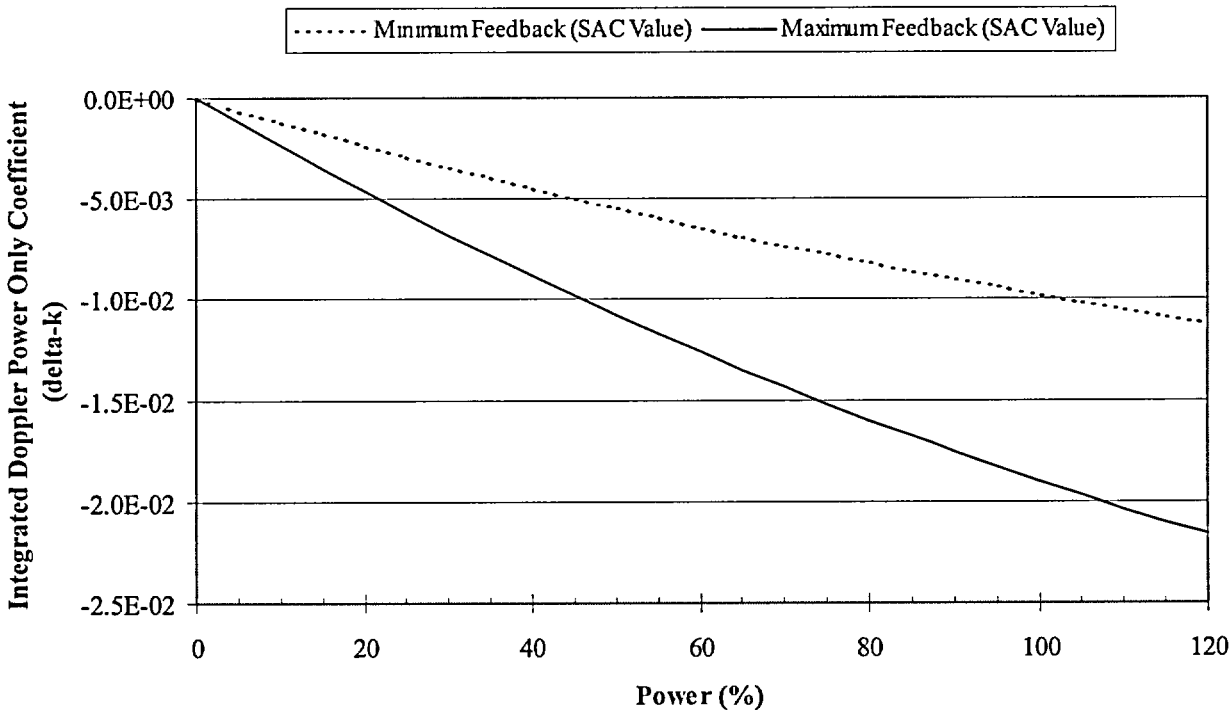


Figure 5.1-6 Integrated DPC Used in Accident Analysis

5.1.1 Uncontrolled RCCA Withdrawal from a Subcritical Condition (USAR Section 14.1.1)

Accident Description

The RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA banks resulting in a power excursion. While the occurrence of a transient of this type is unlikely, such a transient could be caused by a malfunction of the reactor control or the control rod drive system. This could occur with the reactor either subcritical, at hot zero power (HZP), or at power. The "at power" case is discussed in Section 5.1.2.

Withdrawal of an RCCA bank adds reactivity at a prescribed and controlled rate to bring the reactor from a subcritical condition to a low power level during startup. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA bank withdrawal. An RCCA bank movement can cause much faster changes in reactivity than can be made by changing boron concentration (see Section 5.1.4, Chemical and Volume Control System Malfunction).

The RCCA drive mechanisms are wired into pre-selected bank configurations that are not altered during core life. These circuits prevent RCCAs from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled so that no more than two banks can be withdrawn at any time and in their proper withdrawal sequence. The RCCA drive mechanisms are the magnetic latch type; coil actuation is sequenced to provide variable speed travel. The analysis of the maximum reactivity insertion rate includes the assumption of the simultaneous withdrawal of the two sequential banks having the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous control rod assembly withdrawal event occur, the following automatic features of the reactor protection system are available to terminate the transient:

- The source-range high neutron flux reactor trip is actuated when either of two independent source-range channels indicates a neutron flux level above a pre-selected manually adjustable setpoint and provides primary protection below the P6 permissive. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above P6. It is automatically reinstated when both intermediate-range channels indicate a flux level below P6.

- The intermediate range high neutron flux reactor trip is actuated when either of two independent intermediate-range channels indicates a flux level above a pre-selected manually adjustable setpoint. This trip function may be manually bypassed when two-out-of-four power-range channels give readings above the P10 permissive (approximately 10 percent of full power) and is automatically reinstated when three-out-of-four channels indicate a power below P10.

- The power-range high neutron flux reactor trip (low setting) is actuated when two-out-of-four power-range channels indicate a power level above approximately 25 percent of full power. This

trip function may be manually bypassed when two-out-of-four power-range channels indicate a power level above the P10 permissive and is automatically reinstated when three-out-of-four channels indicate a power level below P10.

- The power-range high neutron flux reactor trip (high setting) is actuated when two-out-of-four power-range channels indicate a power level above a preset setpoint (typically, 109-percent power). This trip function is always active while the reactor is at power.

In addition, control rod stops on high intermediate range flux (one-out-of-two) and high power-range flux (one-out-of-four) serve to cease rod withdrawal and prevent the need to actuate the intermediate-range flux trip and the power-range flux trip, respectively.

Method of Analysis

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages. First, a spatial neutron kinetics computer code, TWINKLE, is used to calculate the core average nuclear power transient, including the various core feedback effects, that is, Doppler and moderator reactivity. FACTRAN uses the average nuclear power calculated by TWINKLE and performs a fuel rod transient heat transfer calculation to determine the average heat flux and temperature transients. Finally, the peak core-average heat flux calculated by FACTRAN is used in VIPRE for transient DNBR calculations.

In order to give conservative results for a startup accident, the following assumptions are made:

a. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the DPC, a conservatively low (absolute magnitude) value for the DPC is used (1100 pcm).

b. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the neutron flux response time constant. However, after the initial neutron flux peak, the moderator temperature coefficient (MTC) can affect the succeeding rate of power increase. The effect of moderator temperature changes on the rate of nuclear power increase is calculated in TWINKLE based on temperature-dependent moderator cross-sections. The MTC value used in this event analysis is + 5 pcm/°F.

c. The analysis assumes the reactor to be at HZP nominal temperature of 547°F. This assumption is more conservative than that of a lower initial system temperature (that is, shutdown conditions). The higher initial system temperature yields a larger fuel-to-water heat transfer coefficient, a larger specific heat of the water and fuel, and a less negative (smaller absolute magnitude) DPC. The less negative DPC reduces the Doppler feedback effect, thereby increasing the neutron flux peak. The high neutron flux peak combined with a high fuel-specific heat and larger heat transfer coefficient yields a larger peak heat flux. The analysis assumes the initial effective multiplication factor (K_{eff}) to be 1.0 since this results in the maximum neutron flux peak.

d. Reactor trip is assumed to be initiated by power-range high neutron flux (low setting). The most adverse combination of instrumentation and setpoint errors is accounted for by assuming a 10-percent increase in the power-range flux trip setpoint (low setting), raising it from the nominal value of 25 percent to a value of 35 percent. Figure 5.1.1-1 shows that the rise in nuclear flux is so rapid that the effect of error in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the total reactor trip reactivity is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. Further, the delays for trip signal actuation and control rod assembly release are accounted for in the reactor trip delay time was shown in Table 5.1-5.

e. The maximum positive reactivity insertion rate assumed (100 pcm/in) is a plant-specific value confirmed for each reload cycle and is equal to that for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at a conservative speed (45 in/min, which corresponds to 72 steps/min). It should be noted that the assumption of 72 steps/min as the maximum rod withdrawal speed is contingent upon the performance of refueling interval surveillances as recommended in NSAL-01-001.

f. The DNB analysis assumes the most-limiting axial and radial power shapes possible during the fuel cycle associated with having the two highest combined worth banks in their high worth position.

g. The analysis assumes the initial power level to be below the power level expected for any shutdown condition (10^{-9} fraction of nominal power). The combination of highest reactivity insertion rate and low initial power produces the highest peak heat flux.

h. The analysis is performed at HZP conditions with one reactor coolant pump (RCP) in operation and bounds this accident in lower modes. This assumption also minimizes the resulting DNBR.

i. The accident analysis employs the STDP methodology. Use of the STDP stipulates that the RCS flow rate will be based on the Thermal Design Procedure (TDF) and that the RCS pressure is the nominal pressure minus the uncertainty. Since the event is analyzed from HZP, the steady-state STDP uncertainties on core power and RCS average temperature are not considered in defining the initial conditions.

j. A core flow reduction of 1.1 percent, which addresses the potential reactor coolant flow asymmetry associated with a maximum loop-to-loop steam generator tube plugging (SGTP) imbalance of 10 percent, has been applied.

k. The fuel rod heat transfer calculations performed to determine temperature transients during this event assume a total peaking factor or hot channel factor, F_Q, that is a function of the axial and radial power distributions. The conservatively high value used in this analysis is presented in Table 5.1.1-1.

1. Both Framatome Heavy and Westinghouse 422V+ fuel types, with up to 8 w/o Gadolinia content, were considered in the transient analysis and the most bounding transient results are reported here.

Results

Figures 5.1.1-1 through 5.1.1-5 show the transient behavior for a reactivity insertion rate of 75 pcm/sec, with the accident terminated by the reactor trip at 35 percent of nominal power. The rate is greater than that calculated for the two highest worth sequential control banks, with both assumed to be in their highest incremental worth region.

Figure 5.1.1-1 shows the neutron flux transient. The neutron flux overshoots the full power nominal value for a very short period of time. Therefore, the energy release and fuel temperature increase are relatively small. The heat flux response of interest for the DNB considerations is shown in Figure 5.1.1-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux of much less than the nominal full power value. Figures 5.1.1-3 through 5.1.1-5 show the transient response of the hot spot fuel centerline, fuel average, and cladding temperatures, respectively. Transient DNBR calculations indicate that the minimum DNBR remains above the safety analysis limit value at all times.

Table 5.1.1-1 presents the assumptions and results of the analysis. Table 5.1.1-2 presents the calculated sequence of events. After reactor trip, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal shutdown procedures.

Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and coolant temperature result in a DNBR greater than the limit value. Therefore, no fuel or cladding damage is predicted as a result of this transient.

Table 5.1.1-1 Assumptions and Results – Uncontrolled RCCA Withdrawal from a Subcritical Condition		
Initial Power Level, %	0	
Reactivity Insertion Rate, pcm/sec	75	
Delayed Neutron Fraction	0.0072	
Doppler Power Defect, pcm	1100	
Trip Reactivity, % Δk	1.0	
Hot Channel Factor	6.64	
Number of RCPs Operating	1	
Results		
	Calculated Value	**Limit**
Peak Fuel Centerline Temperature, °F	2685	4746
Peak Fuel Average Temperature, °F	2159	4746
Minimum DNBR (thimble cell)	1.588	1.39
Minimum DNBR (typical cell)	1.733	1.39

Table 5.1.1-2 Sequence of Events – Uncontrolled RCCA Withdrawal from a Subcritical Condition	
Event	**Time (seconds)**
Initiation of Uncontrolled RCCA Bank Withdrawal	0
Power-Range High Neutron Flux Low Setpoint Reached	10.0
Peak Nuclear Power Occurs	10.1
Rod Motion Begins	10.65
Peak Cladding Temperature Occurs	12.3
Peak Heat Flux Occurs	12.4
Minimum DNBR Occurs	12.4
Peak Fuel Average Temperature Occurs	13.1
Peak Fuel Centerline Temperature Occurs	15.2

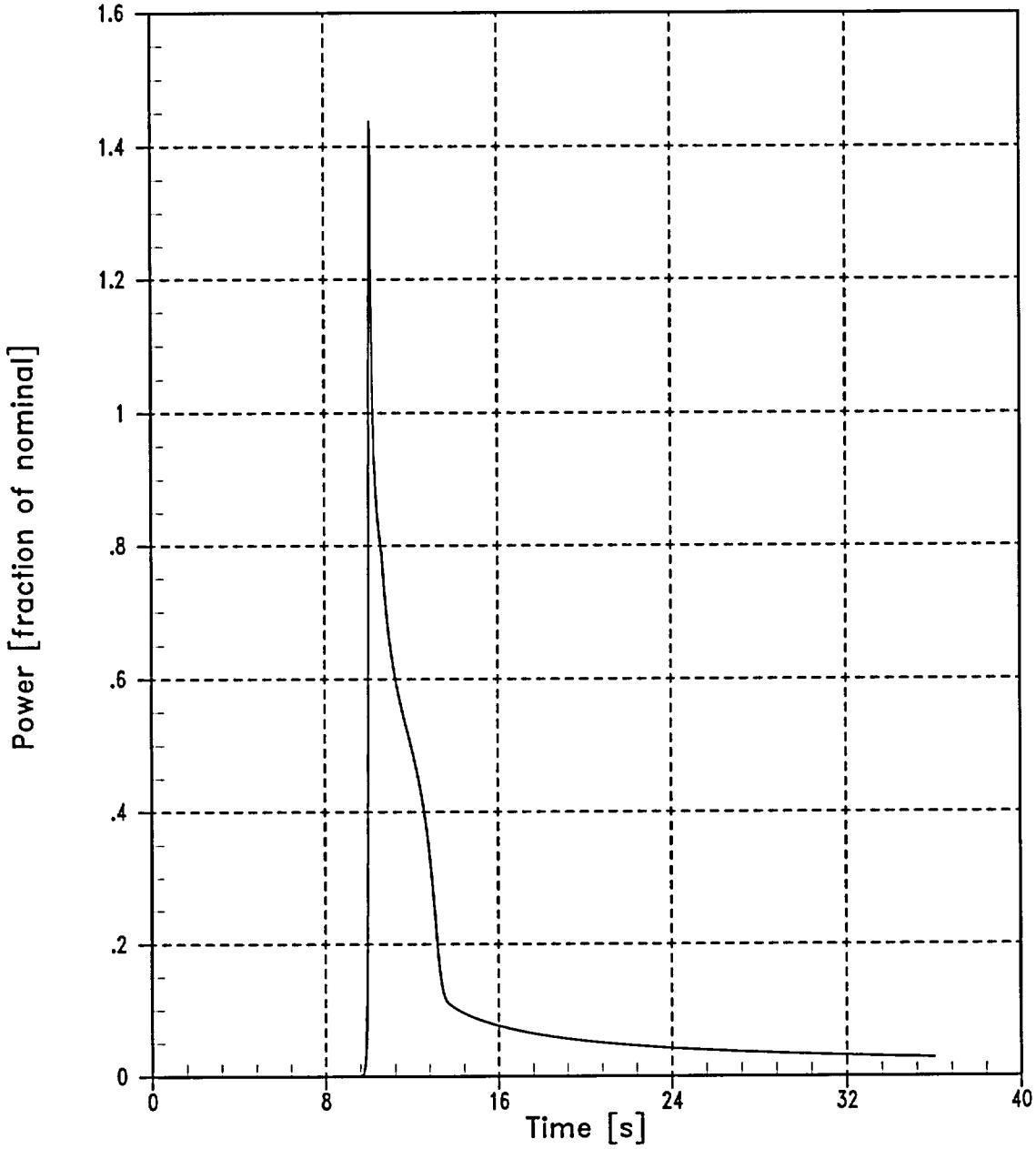


Figure 5.1.1-1 Uncontrolled RCCA Withdrawal from a Subcritical Condition – Reactor Power versus Time

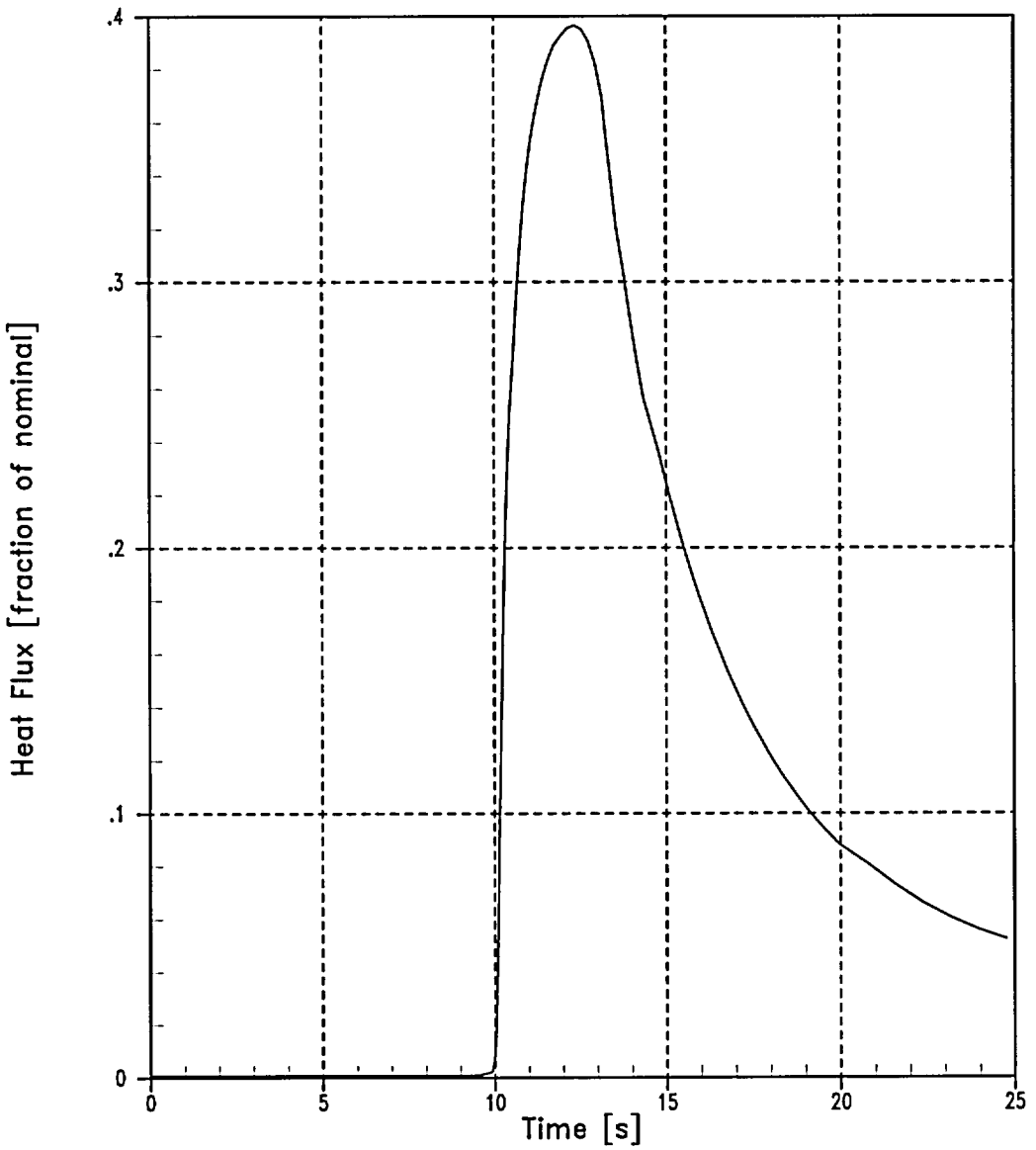


Figure 5.1.1-2 Uncontrolled RCCA Withdrawal from a Subcritical Condition – Heat Flux versus Time

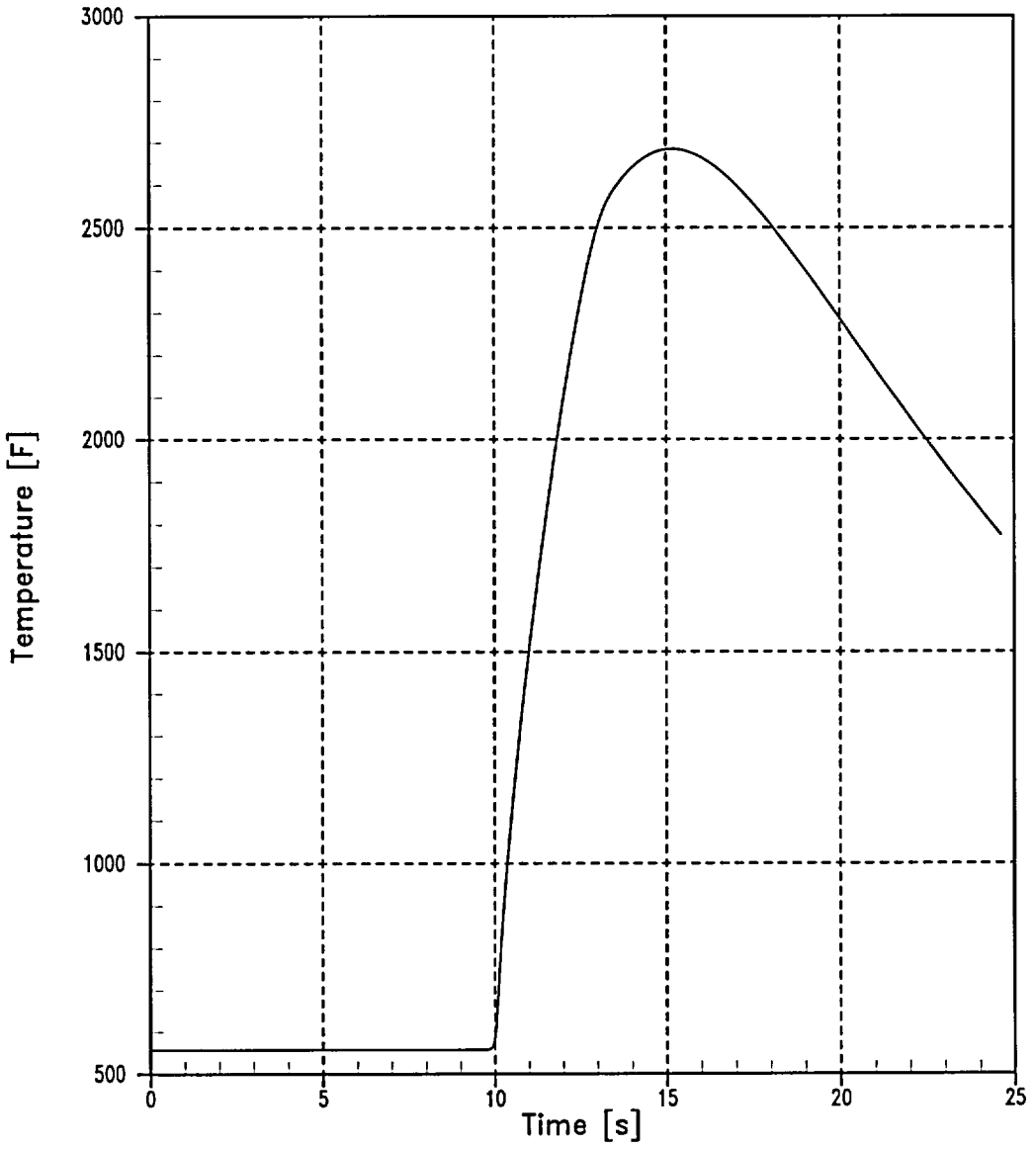


Figure 5.1.1-3 Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Fuel Centerline Temperature versus Time

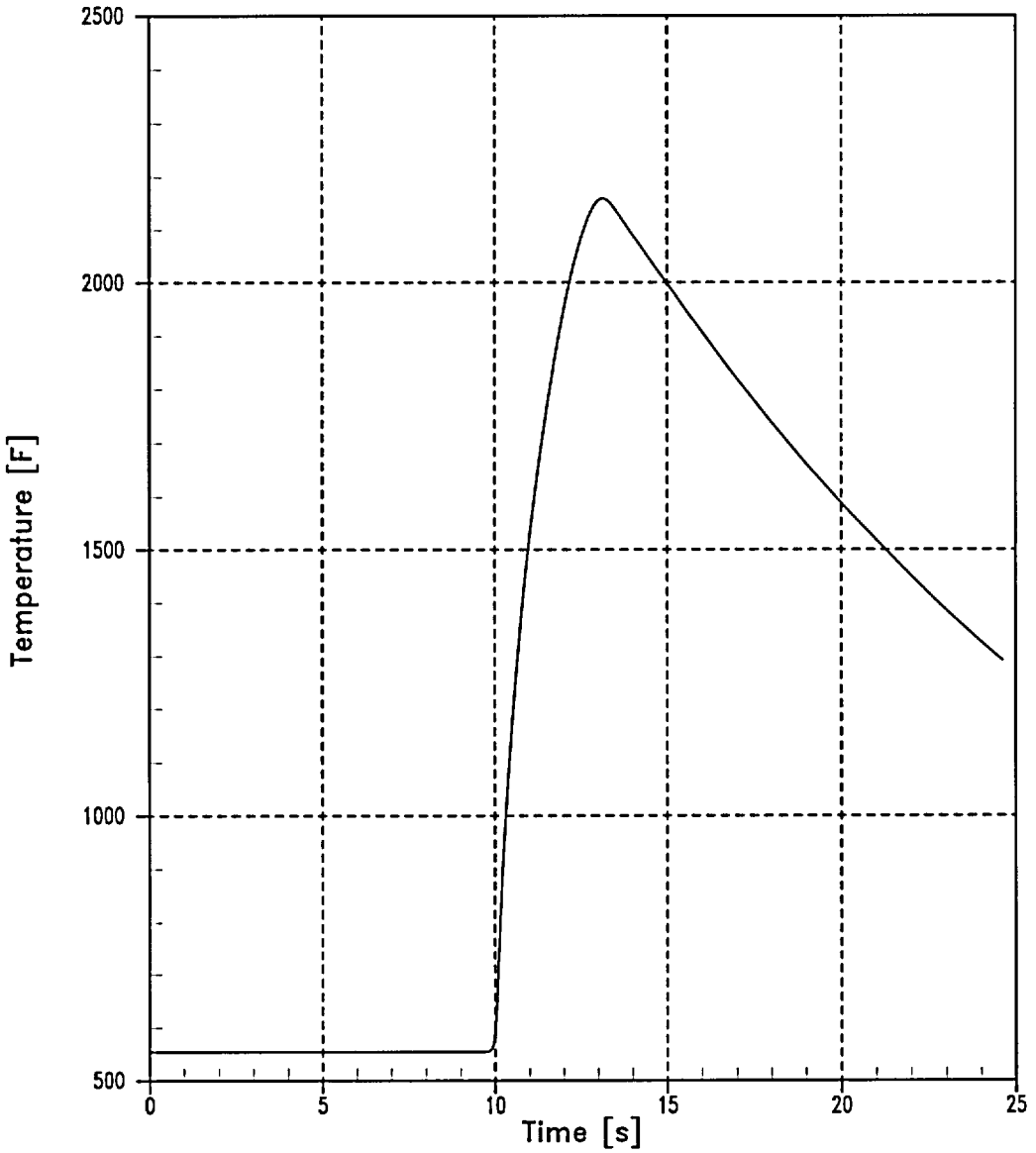


Figure 5.1.1-4 Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Fuel Average Temperature versus Time

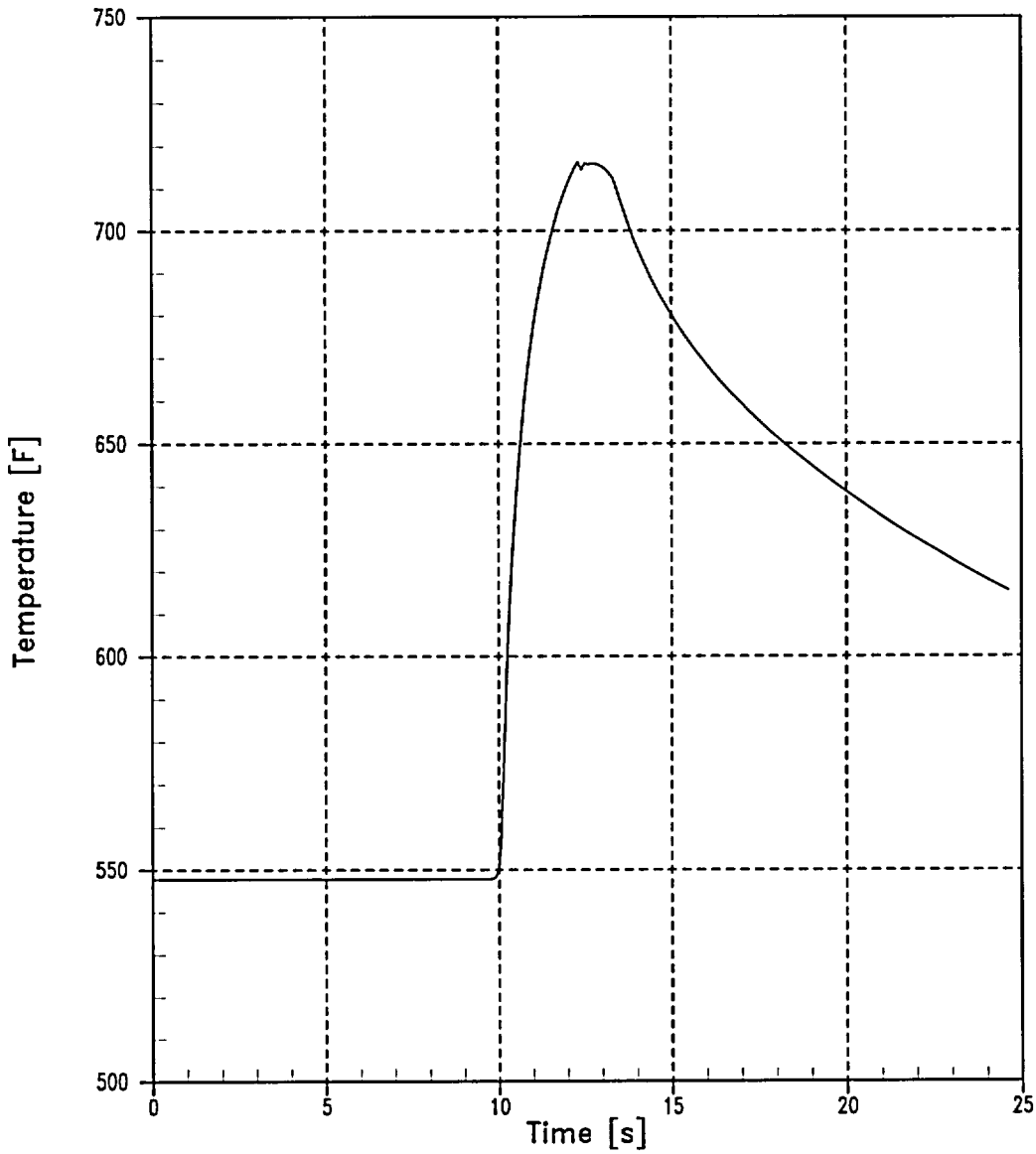


Figure 5.1.1-5 Uncontrolled RCCA Withdrawal from a Subcritical Condition – Hot-Spot Cladding Temperature versus Time

5.1.2 Uncontrolled RCCA Withdrawal at Power (USAR Section 14.1.2)

Accident Description

The uncontrolled RCCA bank withdrawal at power event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of RCCA banks when the core is above the no-load condition. The reactivity insertion resulting from the bank (or banks) withdrawal will cause an increase in core nuclear power and subsequent increase in core heat flux. An RCCA bank withdrawal can occur with the reactor subcritical, at HZP, or at power. The uncontrolled RCCA bank at power event is analyzed for Mode 1 (power operation). The uncontrolled RCCA bank withdrawal from a subcritical or low-power condition is considered as an independent event in Section 5.1.1.

The event is simulated by modeling a constant rate of reactivity insertion starting at time zero and continuing until a reactor trip occurs. The analysis assumes a spectrum of possible reactivity insertion rates up to a maximum positive reactivity insertion rate greater than that occurring with the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum differential rod worth.

Unless the transient RCS response to the RCCA bank withdrawal event is terminated by manual or automatic action, the power mismatch and resultant temperature rise could eventually result in DNB and/or fuel centerline melt. Additionally, the increase in RCS temperature caused by this event will increase the RCS pressure, and if left unchecked, could challenge the integrity of the RCS pressure boundary or the main steam system (MSS) pressure boundary.

To avert the core damage that might otherwise result from this event, the RPS is designed to automatically terminate any such event before the DNBR falls below the limit value, the fuel rod kW/ft limit is reached, the peak pressures exceed their respective limits, or the pressurizer fills. Depending on the initial power level and the rate of reactivity insertion, the reactor may be tripped and the RCCA withdrawal terminated by any of the following trip signals:

- Power-range high neutron flux
- Positive flux rate
- OTΔT
- OPΔT
- High pressurizer pressure
- High pressurizer water level

In addition to the previously listed reactor trips, there are the following withdrawal blocks for the control rod assemblies:

- High nuclear power (one-out-of-four channels)
- High OPΔT (two-out-of-four channels)
- High OTΔT (two-out-of-four channels)

Method of Analysis

The uncontrolled RCCA bank withdrawal at power event is analyzed to show that: (1) the integrity of the core is maintained by the RPS because the DNBR and peak kW/ft remain within the safety analysis limit values and (2) the peak RCS and MS system pressures remain below 110 percent of the corresponding design limits. Of these, the primary concern for this event is assuring that the DNBR limit is met.

The RCCA bank withdrawal at power transient is analyzed with the RETRAN computer program (Reference 5-5). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators, and steam generator relief and safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

To obtain a conservative value for the minimum DNBR, the following analysis assumptions are made:

a. This accident is analyzed with the RTDP (Reference 5-1). Therefore, initial reactor power, pressure, and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR.

b. Reactivity coefficients - Two cases are analyzed:

   1. Minimum reactivity feedback - A zero MTC of reactivity (0 pcm/°F) is assumed at full power. For power levels less than or equal to 60-percent power, a positive MTC of reactivity (+5 pcm/°F) is conservatively assumed, corresponding to the beginning of core life. A conservatively small (in absolute magnitude) DPC is used in the analysis (see Figure 5.1-6).

   2. Maximum reactivity feedback - A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative DPC are assumed (refer to Table 5.1-6 and Figure 5.1-6, respectively).

c. The reactor trip on high neutron flux is actuated at a conservative value of 118 percent of nominal full power. The OTΔT trip includes all adverse instrumentation and setpoint errors. The delays for trip actuation are assumed to be the maximum values (see Note 1 of Table 5.1-5). No credit was taken for the other expected trip functions.

d. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position (see Figure 5.1-5).

e. A range of reactivity insertion rates is examined. The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined differential rod worth at a conservative speed (45 inches/minute, which corresponds to 72 steps/minute).

f. Power levels of 10 percent, 60 percent and 100 percent of full power are considered.

g. The impact of a full-power RCS vessel T_{avg} window was considered for the uncontrolled RCCA bank withdrawal at power analysis. A conservative calculation modeling the high end of the RCS vessel T_{avg} window was explicitly analyzed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in the OTΔT trip setpoint proportional to a decrease in margin to DNB. A separate case analyzing the RCCA bank withdrawal at power transient for maximum RCS pressure is documented in the analysis calculation note (see Table 5.1-8).

Results

The limiting results were calculated for the RCCA bank withdrawal at power transient analyzed for the FU/PU implementation. They are given in Table 5.1.2-2.

RCS pressures below the limit of 2750 psia are obtained for reactivity insertion rates less than or equal to 84 pcm/second. This reactivity insertion rate bounds that calculated for the simultaneous withdrawal, at maximum speed, of two sequential RCCA banks having the maximum differential rod worth.

Figure 5.1.2-1 shows the response of nuclear power, pressurizer pressure, RCS vessel T_{avg}, and DNBR to a rapid RCCA withdrawal incident starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in reactor core T_{avg} and pressurizer pressure result, and a large margin to DNB is maintained.

The response of nuclear power, pressurizer pressure, RCS vessel T_{avg}, and DNBR for a slow control rod assembly withdrawal from 100-percent power is shown in Figure 5.1.2-2. Reactor trip on OTΔT occurs after a longer period of time and the rise in temperature is consequently larger than for a rapid RCCA withdrawal. Again, the minimum DNBR is greater than the limit value.

Figure 5.1.2-3 shows the minimum DNBR as a function of the reactivity insertion rate for the three initial power levels (100 percent, 60 percent, and 10 percent) and for minimum and maximum reactivity feedback. It can be seen that the high neutron flux and OTΔT trip channels provide protection over the whole range of reactivity insertion rates because the minimum DNBR is never less than the limit value.

In the referenced figures, the shape of the curves of minimum DNBR versus reactivity insertion rate is due both to the reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 5.1.2-3 (sheet 3) for example, it is noted that:

1. For high reactivity insertion rates (that is, between ~100 pcm/second and ~30 pcm/second) when modeling minimum reactivity feedback, reactor trip is initiated by the high neutron flux trip. The neutron flux level in the core rises rapidly for these insertion rates, while core heat flux and coolant temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Therefore, the reactor is tripped prior to a significant increase in the heat flux or core water temperature with resultant high minimum DNBRs during the transient. Within this range, as the

reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Therefore, minimum DNBR during the transient decreases with decreasing reactivity insertion rate.

2. With a further decrease in the reactivity insertion rate, the OTΔT and high neutron flux trips become equally effective in terminating the transient (such as, at a reactivity insertion rate of approximately 30 pcm/second).

   The OTΔT reactor trip function initiates a reactor trip when the measured ΔT exceeds an OTΔT setpoint that is based on the measured vessel T_{avg} and pressurizer pressure. It is important to note, however, that the contribution of RCS vessel T_{avg} to the OTΔT trip function is lead-lag compensated to compensate for the effect of the thermal capacity of the RCS response to power increases.

   For reactivity insertion rates between ~30 pcm/second and ~8 pcm/second, the effectiveness of the OTΔT trip increases (in terms of increased minimum DNBR). This is due to the fact that, with lower insertion rates, the power increase rate is slower, the rate of rise of RCS vessel T_{avg} is slower, and the system lags and delays become less significant.

3. For reactivity insertion rates of ~8 pcm/second and lower, the rise in RCS temperature is sufficiently high so that there is an increased steam relief through the steam generator safety valves prior to trip. This steam relief acts as an additional heat sink on the RCS and sharply slows the increase of the RCS vessel T_{avg}. This causes the OTΔT trip setpoint to be reached later with resulting lower minimum DNBRs.

Conclusions

The results for the uncontrolled RCCA bank withdrawal at power transient analyzed show that the high neutron flux and OTΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; that is, the minimum calculated DNBR is always greater than the safety analysis limit value. In addition analysis results show that the peak kW/ft is less than the limit and the peak pressures in the RCS and secondary steam system do not exceed 110 percent of their respective design pressures.

Thus, all pertinent criteria are met for the uncontrolled RCCA bank withdrawal at power transient when assuming the FU/PU implementation.

Table 5.1.2-1 Time Sequence of Events for Uncontrolled RCCA Withdrawal at Power (Maximum Nominal RCS T_{avg}; Minimum Feedback)	
Event	**Time (Seconds)**
Case A:	
Initiation of Uncontrolled RCCA Withdrawal at Full Power with Minimum Reactivity Feedback (100 pcm/sec)	0
Power-Range High Neutron Flux High Trip Point Reached	1.38
Rods Begin to Fall into Core	2.03
Minimum DNBR Occurs	2.75
Case B:	
Initiation of Uncontrolled RCCA Withdrawal at Full Power with Minimum Reactivity Feedback (3 pcm/sec)	0
OTΔT Reactor Trip Signal Initiated	45.28
Rods Begin to Fall into Core	47.28
Minimum DNBR occurs	47.63

Table 5.1.2-2 Limiting Results for RCCA Bank Withdrawal at Power Transient			
Criterion	**Limiting Value**	**Analysis Limit**	**Case**
DNBR	1.46	1.34	Full power, minimum reactivity feedback, 4 pcm/second reactivity insertion rate
Core Heat Flux (FON)	1.17	1.18	Full power, maximum reactivity feedback 37 pcm/second reactivity insertion rate
MSS Pressure (psia)	1204	1210	60% of full power, maximum reactivity feedback, 5 pcm/second reactivity insertion rate

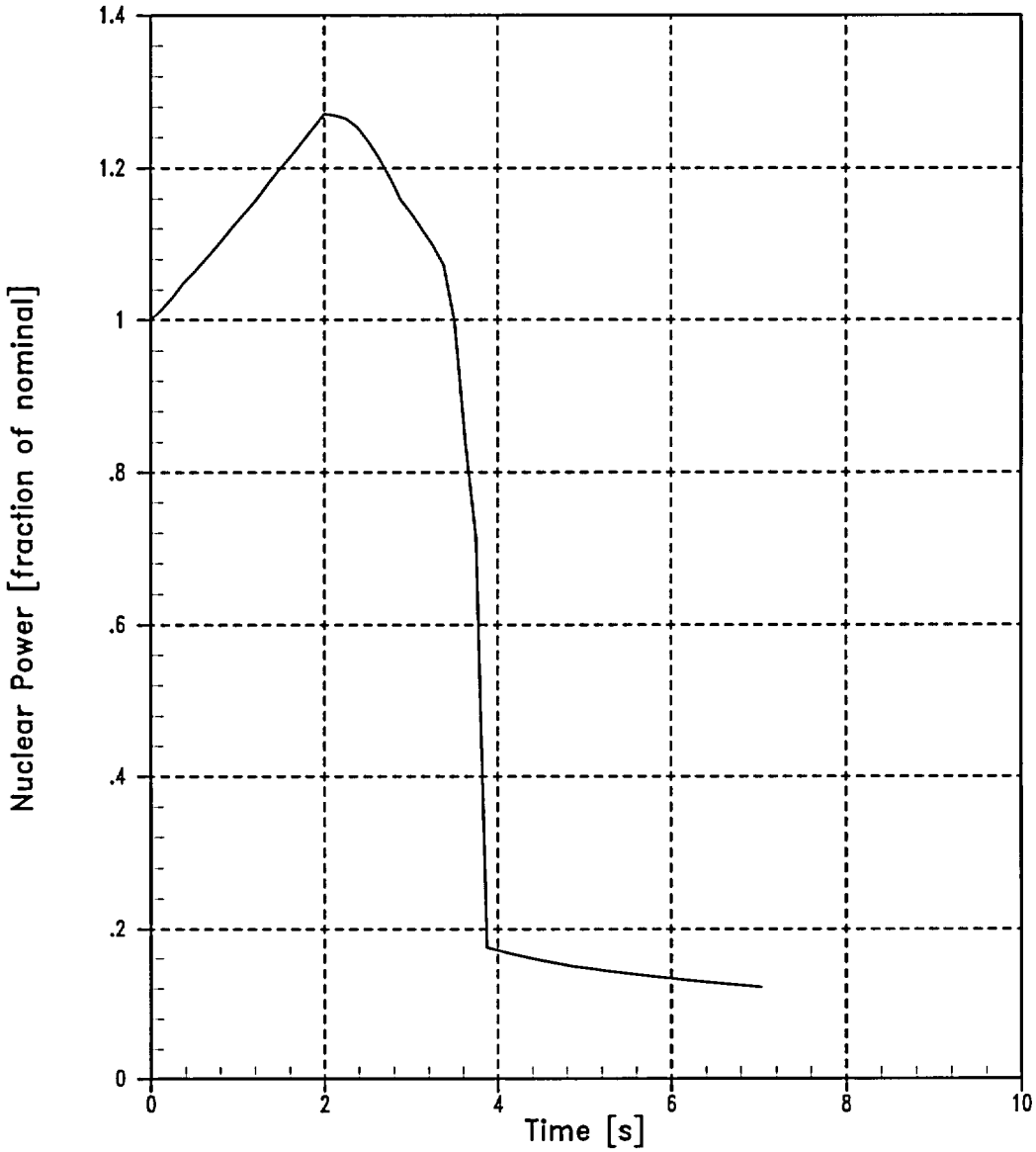


Figure 5.1.2-1 Uncontrolled RCCA Bank Withdrawal at Power (100 pcm/sec - Full Power) Power-Range High Neutron Flux Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 1 of 4)

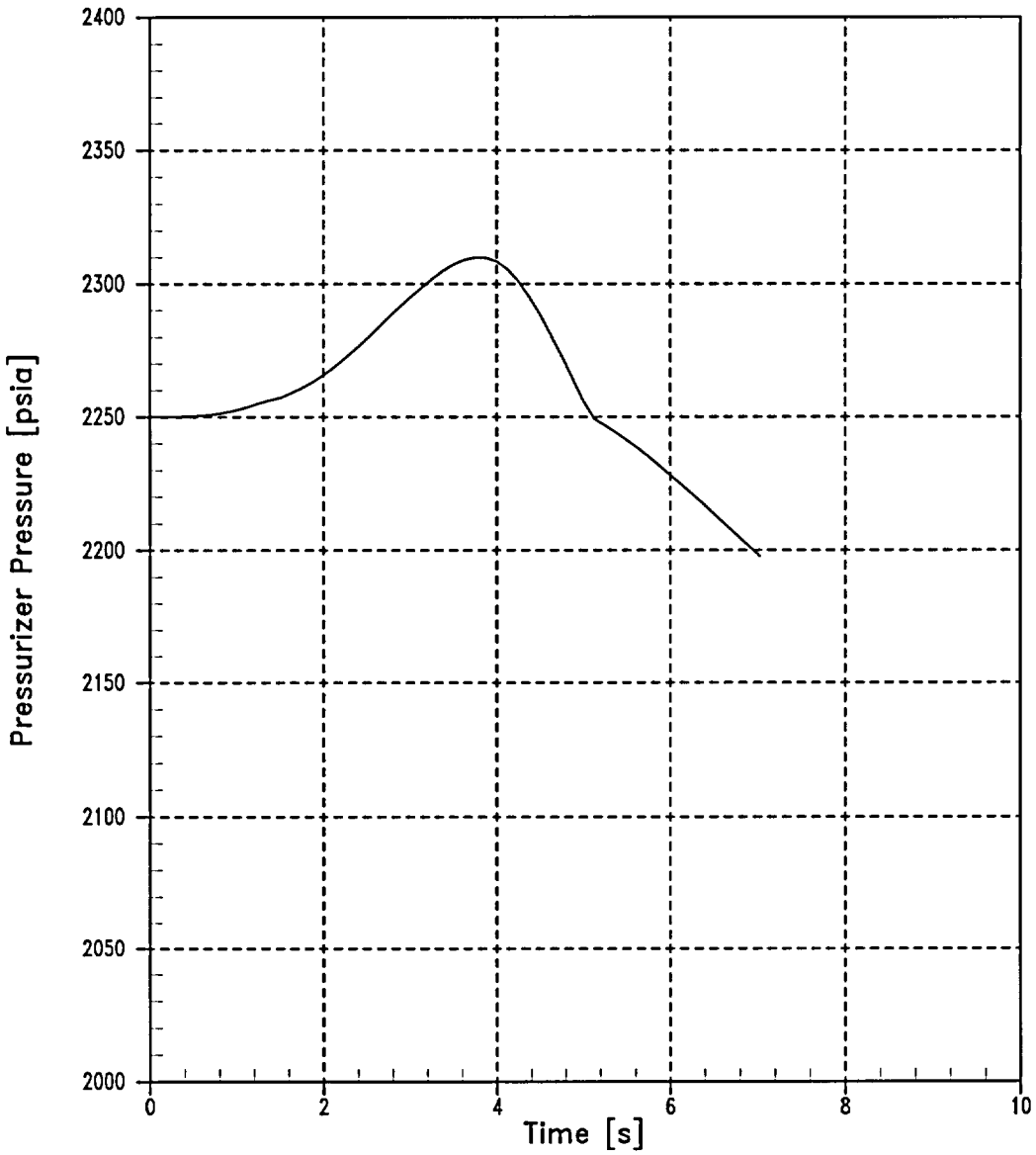


Figure 5.1.2-1 Uncontrolled RCCA Bank Withdrawal at Power (100 pcm/sec - Full Power) Power-Range High Neutron Flux Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 2 of 4)

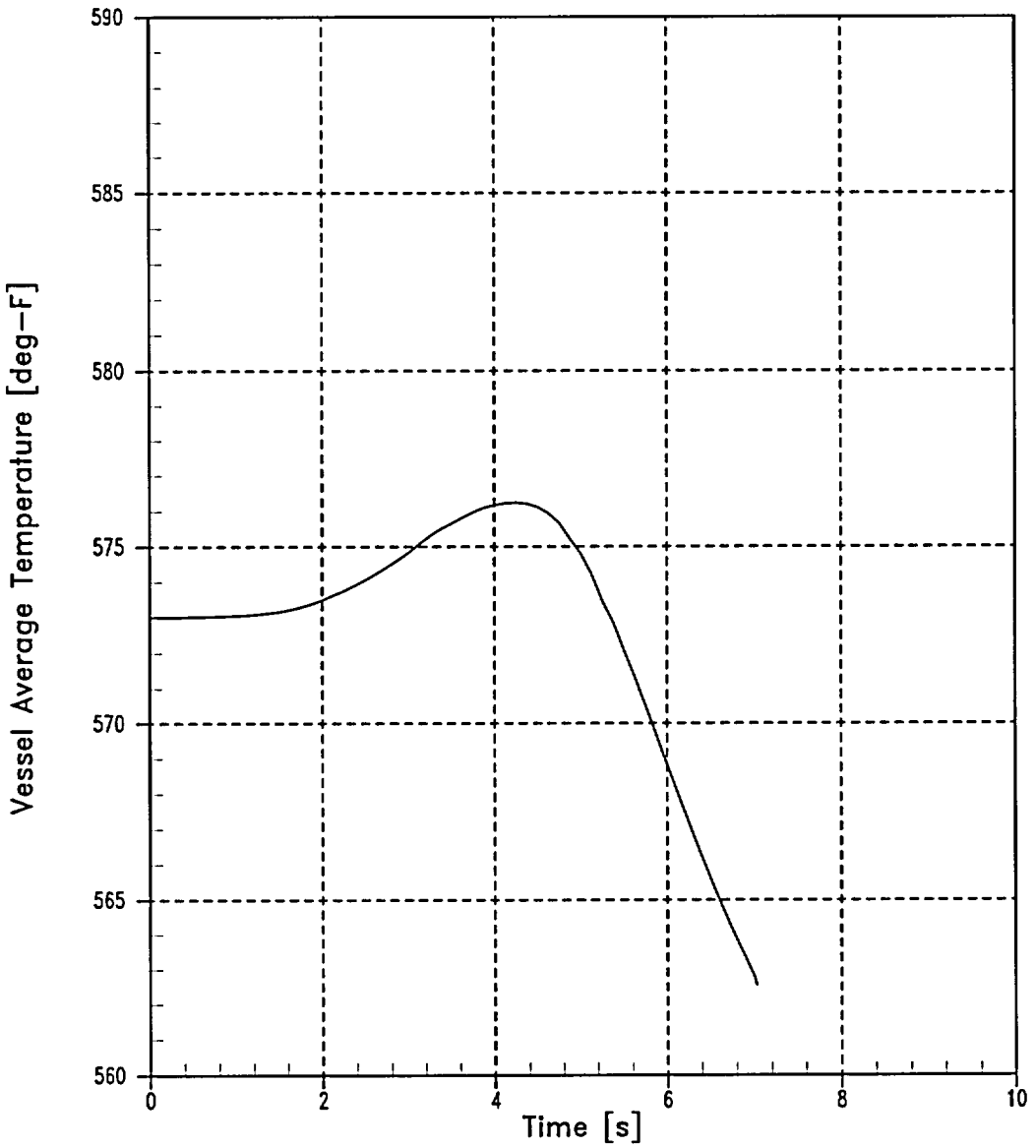


Figure 5.1.2-1 Uncontrolled RCCA Bank Withdrawal at Power (100 pcm/sec - Full Power) Power-Range High Neutron Flux Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 3 of 4)

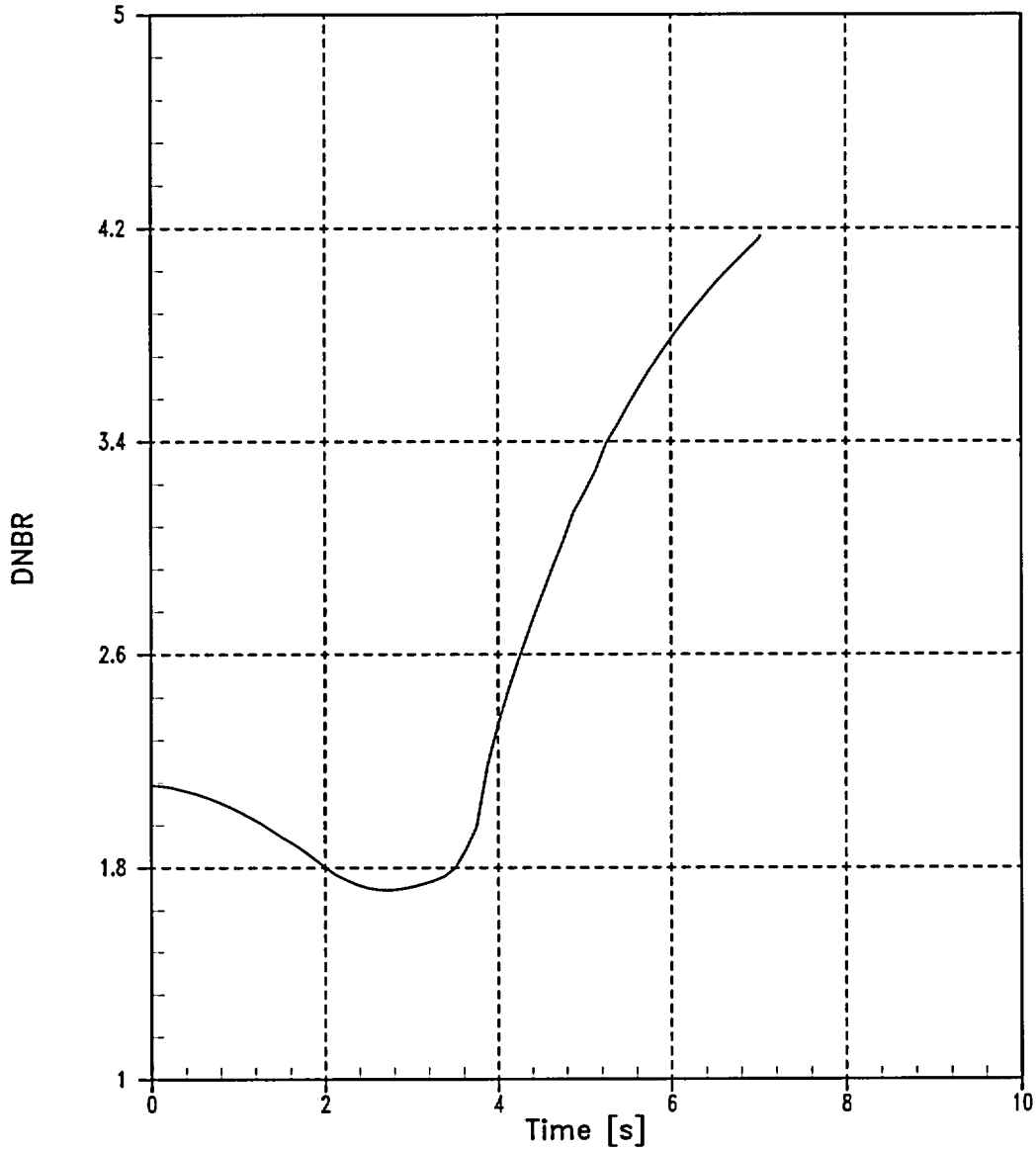


Figure 5.1.2-1 Uncontrolled RCCA Bank Withdrawal at Power (100 pcm/sec - Full Power) Power-Range High Neutron Flux Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 4 of 4)

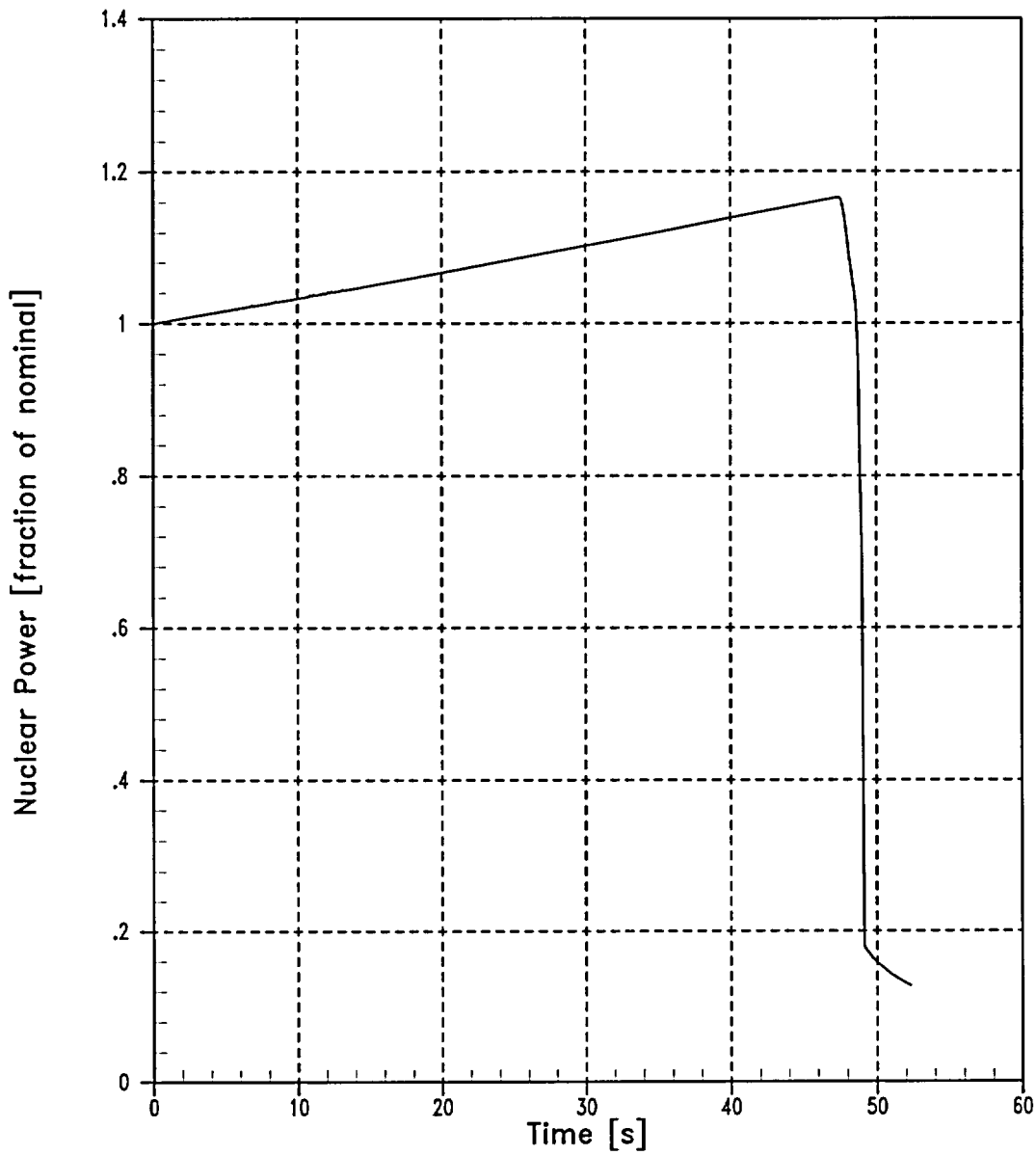


Figure 5.1.2-2 Uncontrolled RCCA Bank Withdrawal at Power (3 pcm/sec - 100% Power) OTΔT Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 1 of 4)

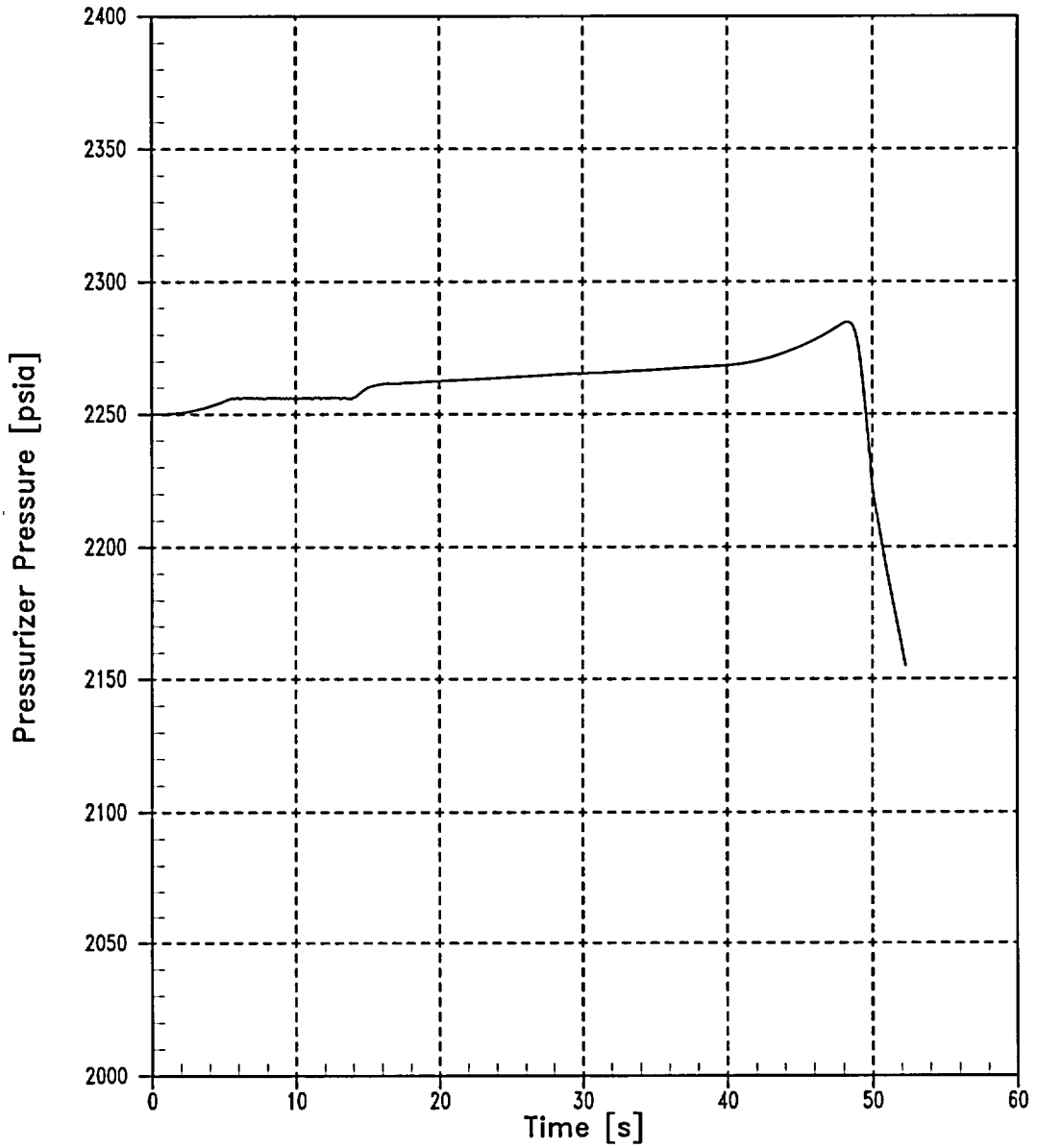


Figure 5.1.2-2 Uncontrolled RCCA Bank Withdrawal at Power (3 pcm/sec - 100% Power) OTΔT Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 2 of 4)

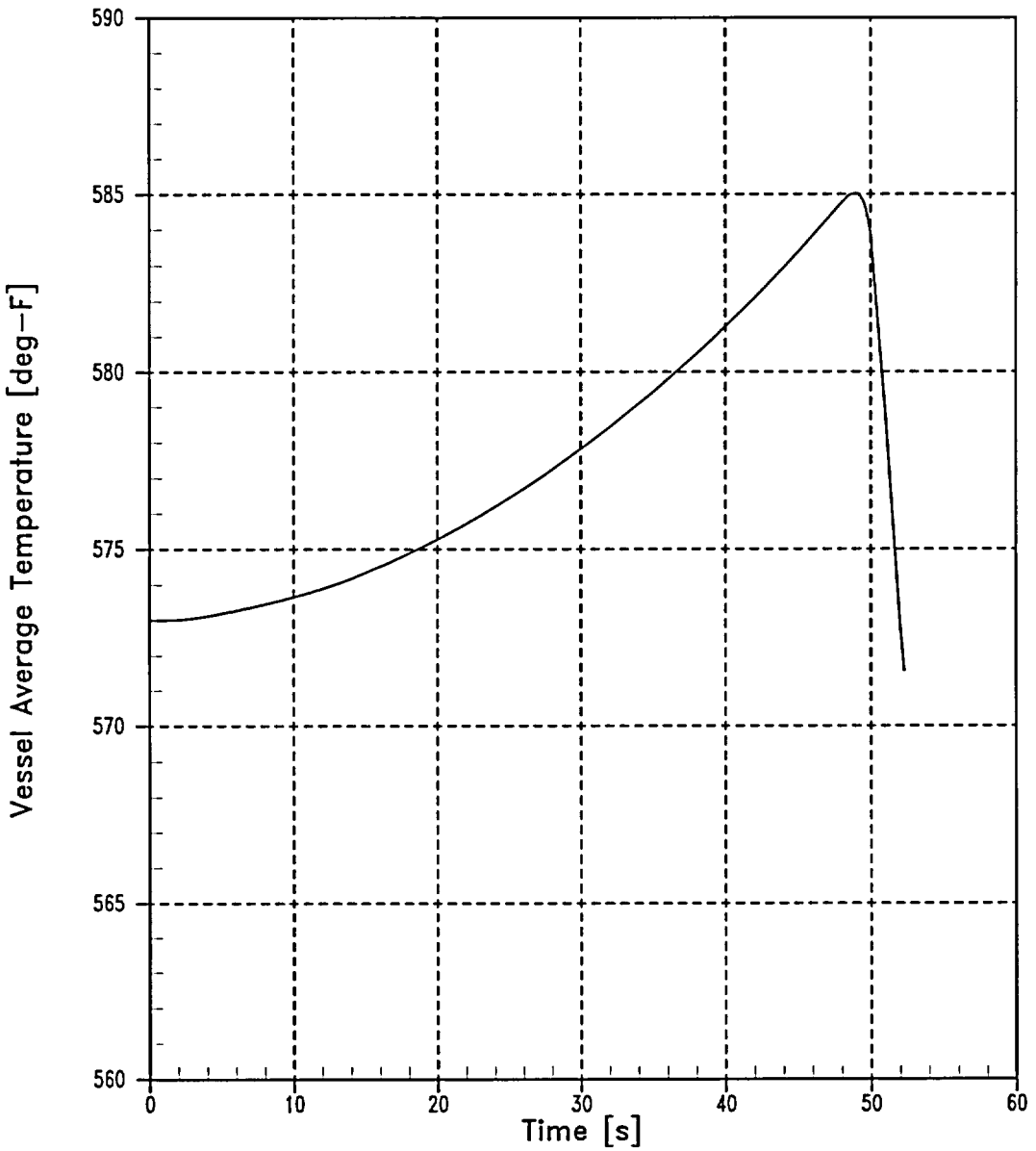


Figure 5.1.2-2 Uncontrolled RCCA Bank Withdrawal at Power (3 pcm/sec - 100% Power) OTΔT Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 3 of 4)

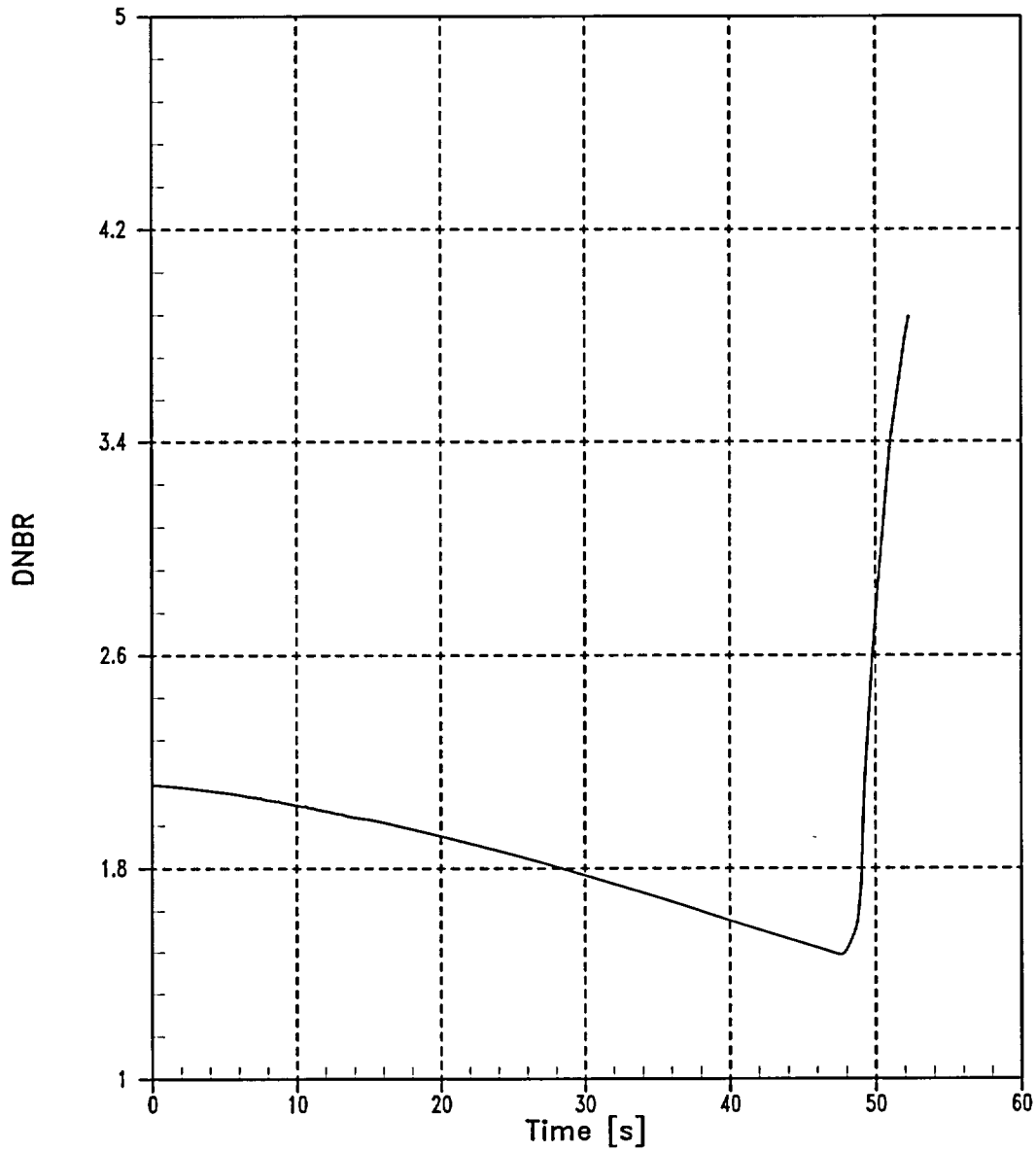


Figure 5.1.2-2 Uncontrolled RCCA Bank Withdrawal at Power (3 pcm/sec - 100% Power) OTΔT Trip, Maximum Nominal RCS Vessel T_{avg}, Minimum Feedback (Sheet 4 of 4)

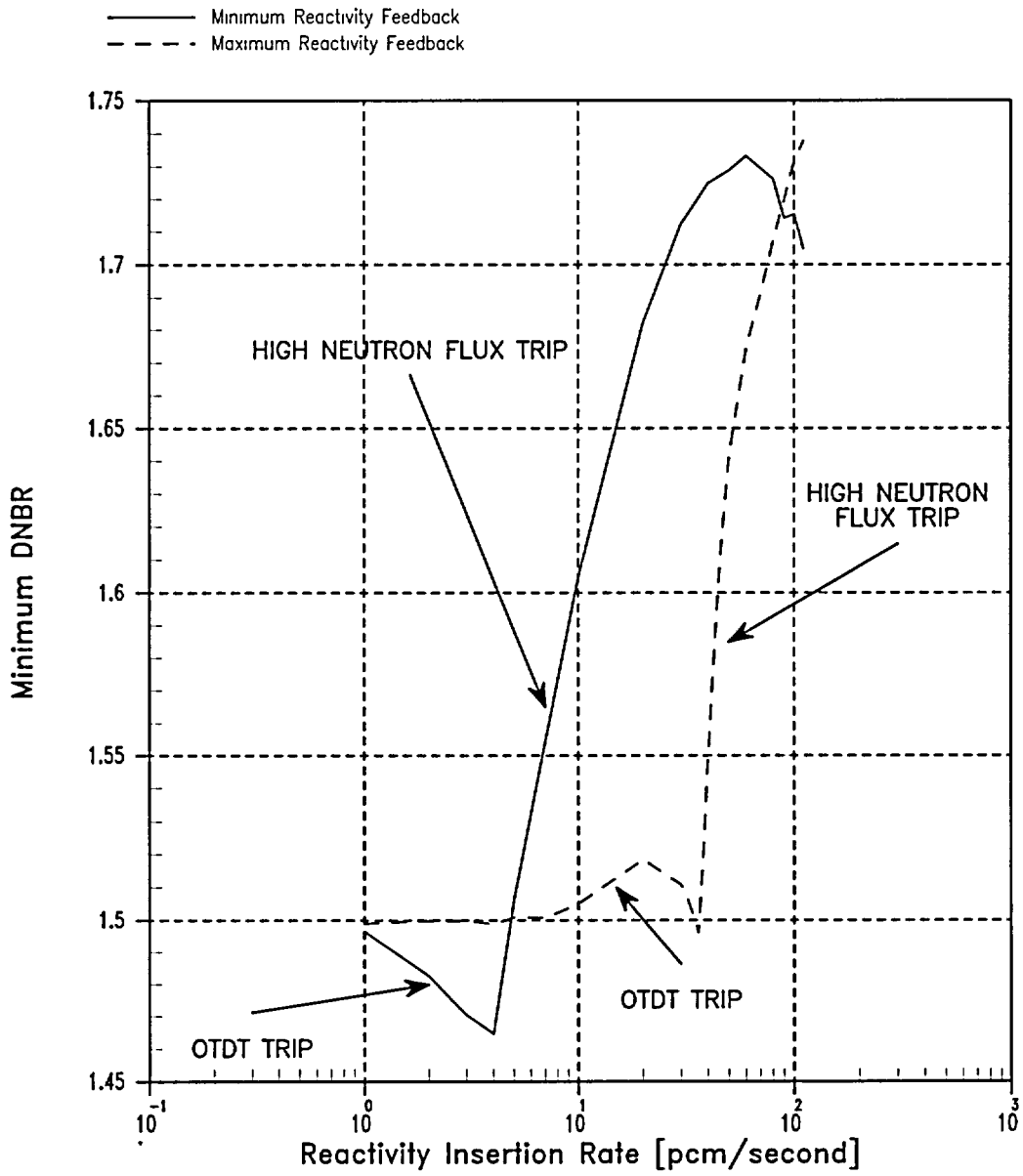


Figure 5.1.2-3 Uncontrolled RCCA Bank Withdrawal at Power, 100% Power (Sheet 1 of 3)

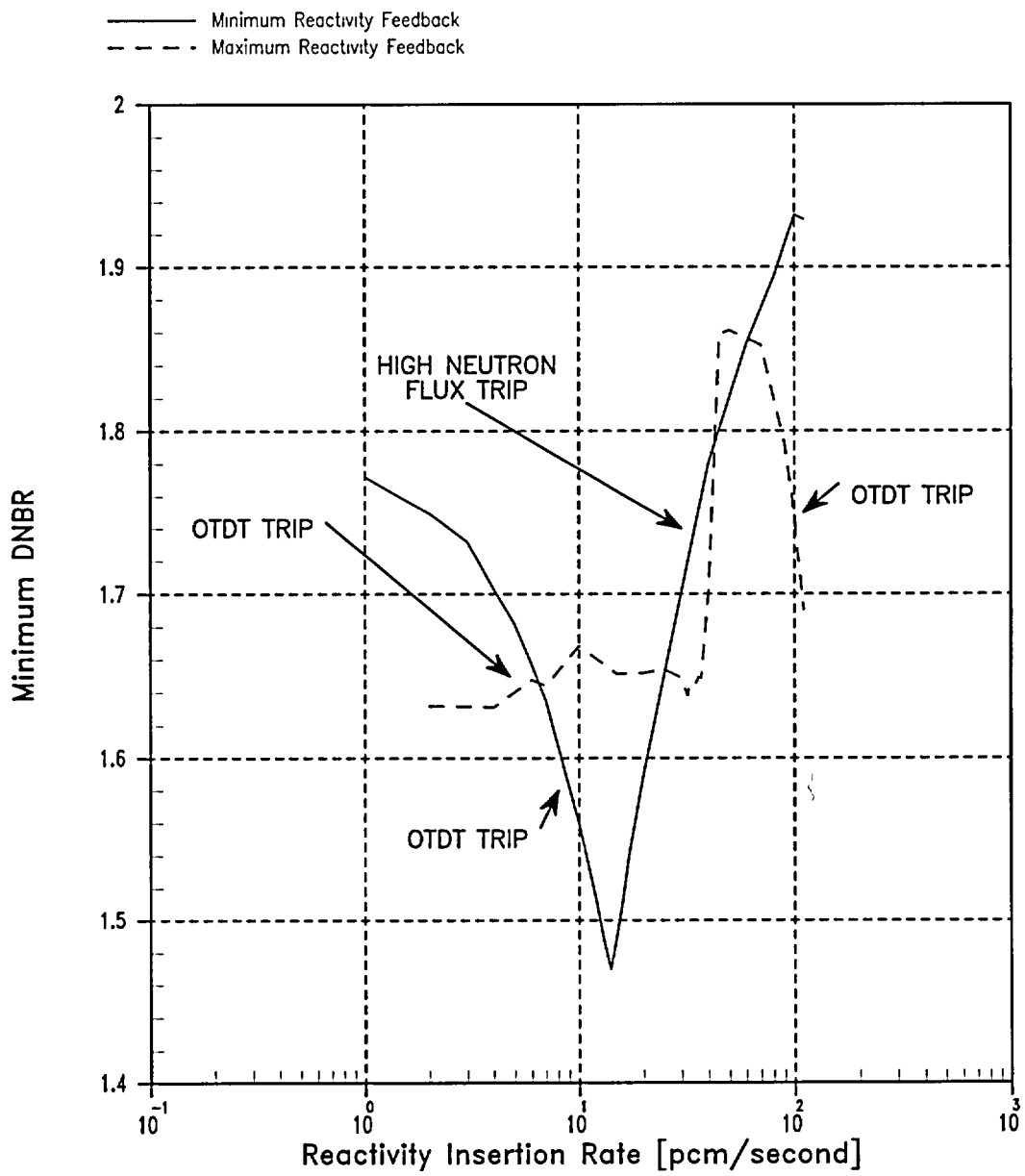


Figure 5.1.2-3 Uncontrolled RCCA Bank Withdrawal at Power, 60% Power (Sheet 2 of 3)

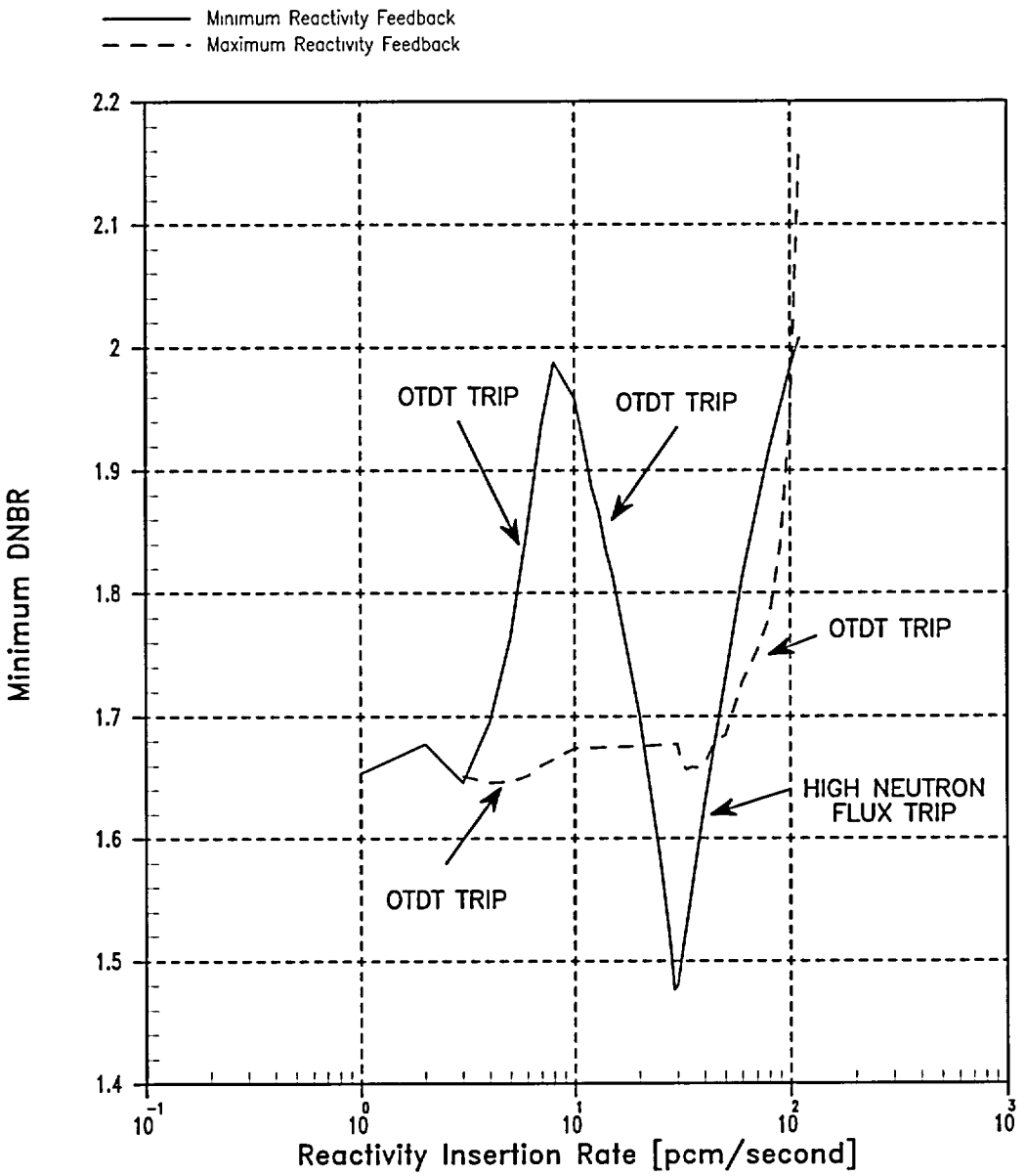


Figure 5.1.2-3 Uncontrolled RCCA Bank Withdrawal at Power, 10% Power (Sheet 3 of 3)

5.1.3 Rod Cluster Control Assembly Misalignment (USAR Section 14.1.3)

Accident Description

The RCCA misalignment accidents include:

- Dropped full-length RCCAs
- Dropped full-length RCCA banks
- Statically misaligned full-length RCCAs

Each RCCA has a rod position indicator channel that displays the position of the assembly. The displays of assembly positions are grouped for operator convenience. Fully inserted assemblies are further indicated by rod bottom lights. The full-length assemblies are always moved in pre-selected banks and the banks are always moved in the same pre-selected sequence.

Dropped assemblies or assembly banks are detected by:

- Sudden drop in the core power level
- Asymmetric power distribution (as seen on out-of-core neutron detectors or core exit thermocouples)
- Rod bottom light(s)
- Rod deviation alarm (if the plant computer is in operation)
- Rod position indicators

Misaligned assemblies are detected by:

- Asymmetric power distribution (as seen on out-of-core neutron detectors or core exit thermocouples)
- Rod deviation alarm (if the plant computer is in operation)
- Rod position indicators

Method of Analysis

One or More Dropped RCCAs from the Same Group

The LOFTRAN computer code calculates transient system responses for the evaluation of a dropped RCCA event. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power levels.

Transient RCS statepoints (temperature, pressure, and power) are calculated by LOFTRAN. Nuclear models are used to obtain a hot-channel factor consistent with the primary-system conditions and reactor power. By incorporating the primary conditions from the transient analysis and the hot-channel factor from the nuclear analysis, it is shown that the DNB design basis is met using the VIPRE code. The transient response analysis, nuclear peaking factor analysis, and performance of the DNB design basis confirmation are performed in accordance with approved the methodology described in Reference 5-12

and in line with the Reference 5-1 guidelines. The analysis does not take credit for the power-range negative flux rate reactor trip.

A generic statepoint analysis for this event, which was performed in 1986 to bound a number of two-loop PWRs, was evaluated and determined to be applicable to KNPP for the FU/PU Program. With the generic statepoints being applicable, the effects of the fuel transition and power uprate are accounted for in the DNB analysis, which is performed on a cycle-specific basis.

Dropped RCCA Bank

A dropped RCCA bank results in a symmetric power change in the core. Assumptions made in the methodology (Reference 5-12) for the dropped RCCA(s) analysis provide a bounding analysis for the dropped RCCA bank.

A generic statepoint analysis for this event, which was performed in 1986 to bound a number of two-loop PWRs, was evaluated and determined to be applicable to KNPP for the FU/PU Program. With the generic statepoints being applicable, the effects of the fuel transition and power uprate are accounted for in the DNB analysis, which is performed on a cycle-specific basis.

Statically Misaligned RCCA

Steady-state power distributions are analyzed using the appropriate nuclear physics computer codes. The peaking factors are then used as input to the VIPRE code to calculate the DNBR. The following cases are examined in the analysis assuming the reactor is initially at full power: the worst rod withdrawn with bank D inserted at the insertion limit, the worst rod dropped with bank D inserted at the insertion limit, and the worst rod dropped with all other rods out. It is assumed that the incident occurs at the time in the cycle at which the maximum all-rods-out $F_{\Delta H}$ occurs. This assures a conservative $F_{\Delta H}$ for the misaligned RCCA configuration.

Results

One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion. The core is not adversely affected during this period since power is decreasing rapidly. Either reactivity feedback or control bank withdrawal will re-establish power.

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. Without control system interaction, a new equilibrium is achieved at a reduced power level and reduced primary temperature. Therefore, the automatic rod control mode of operation is the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller, after which the control system will insert the control bank to restore nominal power. Figures 5.1.3-1 through 5.1.3-4 show a typical transient response to a dropped RCCA (or RCCAs)

event with the reactor in automatic rod control. In all cases, the minimum DNBR remains above the limit value.

Following plant stabilization, the operator may manually retrieve the RCCA(s) by following approved operating procedures.

Dropped RCCA Bank

A dropped RCCA bank results in a negative reactivity insertion greater than 500 pcm. The core is not adversely affected during the insertion period, since power is decreasing rapidly. The transient will proceed similar to that described in the previous "One or More Dropped RCCAs" section, but the return to power will be less due to the greater negative reactivity worth of an entire RCCA bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

Statically Misaligned RCCA

The most severe RCCA misalignment situations with respect to DNB at significant power levels are associated with cases in which one RCCA is fully inserted with either all rods out or bank D at the insertion limit, or where bank D is inserted to the insertion limit and one RCCA is fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the transient approaches the postulated conditions.

The insertion limits in the Technical Specifications may vary from time to time, depending on several limiting criteria. The full-power insertion limits on control bank D must be chosen to be above that position which meets the minimum DNBR and peaking factors. The full-power insertion limit is usually dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For the RCCA misalignment case with one RCCA fully inserted (with either all rods out or bank D at the insertion limit), the DNBR does not fall below the limit value. The analysis for this case assumes that the initial reactor power, RCS pressure, and RCS temperature are at nominal values with uncertainties included, and with the increased radial peaking factor associated with the misaligned RCCA.

For the RCCA misalignment case with bank D inserted to the full-power insertion limit and one RCCA fully withdrawn, the DNBR does not fall below the limit value. The analysis for this case assumes that the initial reactor power, RCS pressure, and RCS temperature are at nominal values with uncertainties included, and with the increased radial peaking factor associated with the misaligned RCCA.

Departure from nucleate boiling does not occur for the RCCA misalignment incident. Therefore, there is no reduction in the ability of the primary coolant to remove heat from the fuel rod. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation rate is well below that which would cause fuel melting.

After identifying an RCCA group misalignment condition, the operator must take action as required by the plant Technical Specifications and operating procedures.

Conclusions

The evaluation of the generic statepoints that were obtained using the methodology in Reference 5-12, for cases of dropped RCCAs or dropped banks encompassing all possible dropped rod worths delineated in Reference 5-12, concluded that the minimum DNBR remains above the safety analysis limit value. For all cases of any single RCCA fully inserted, or bank D inserted to the rod insertion limit and any single RCCA in that bank fully withdrawn (static misalignment), the minimum DNBR remains above the limit value. Therefore, the DNB design criterion is met and the RCCA misalignments do not result in core damage given implementation of the FU/PU Program.

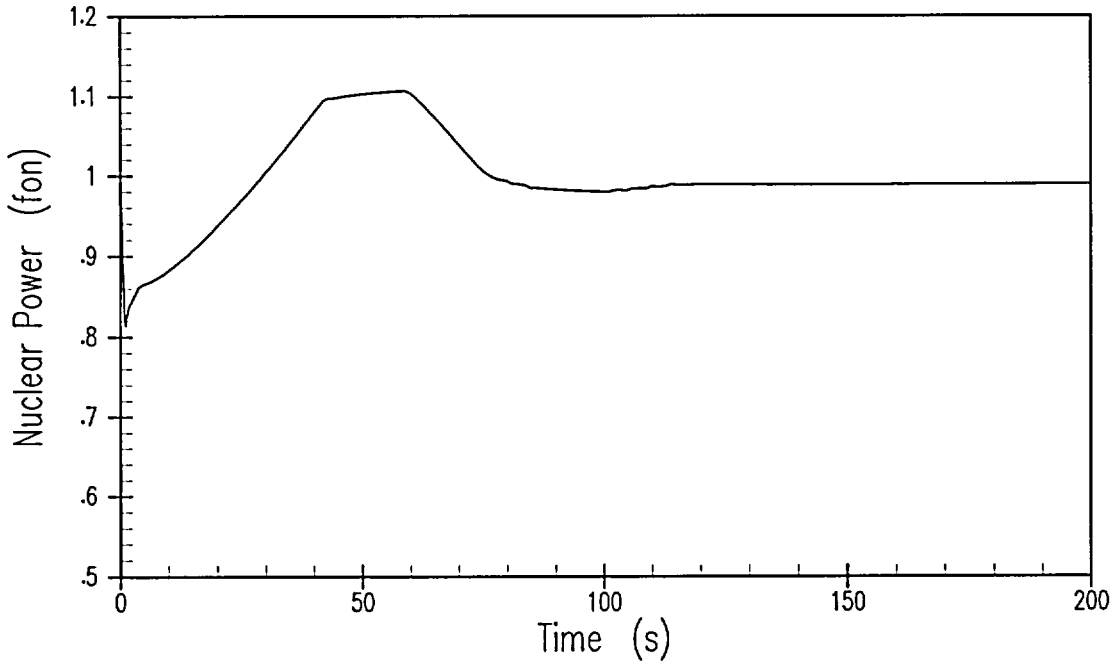


Figure 5.1.3-1 Representative Transient Response to Dropped RCCA – Nuclear Power versus Time

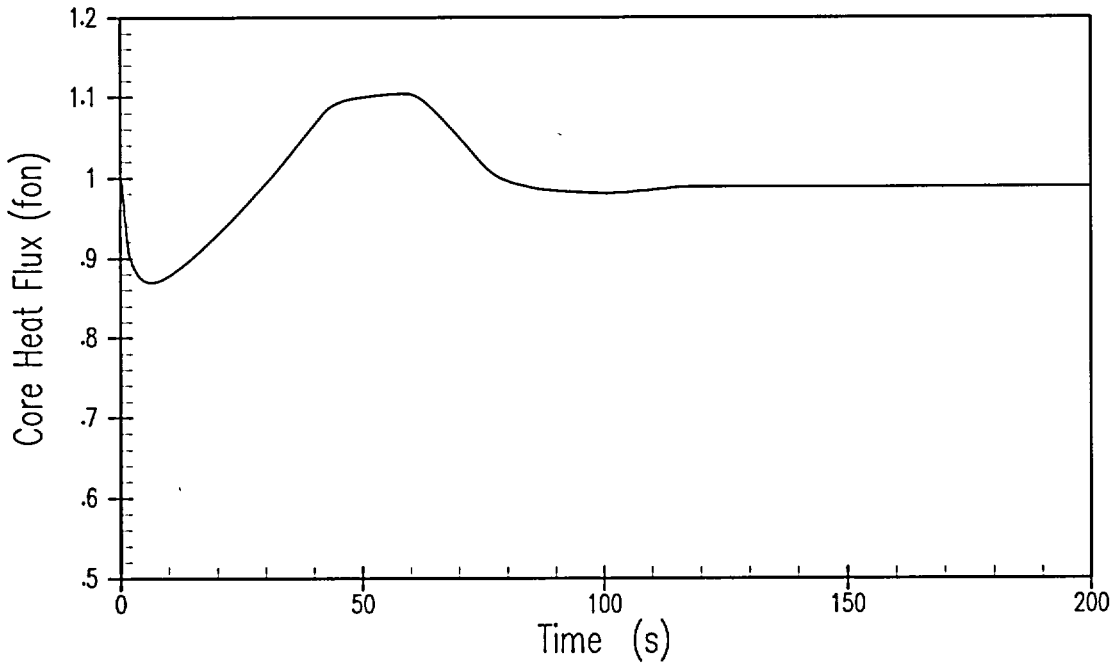


Figure 5.1.3-2 Representative Transient Response to Dropped RCCA – Core Heat Flux versus Time

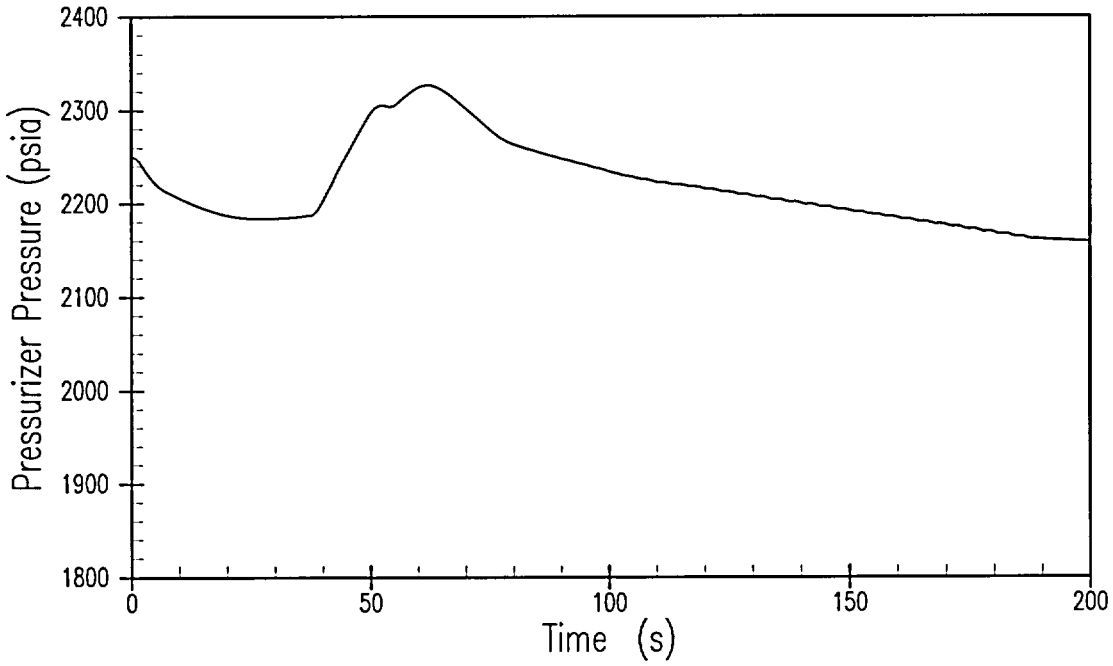


Figure 5.1.3-3 Representative Transient Response to Dropped RCCA – Pressurizer Pressure versus Time

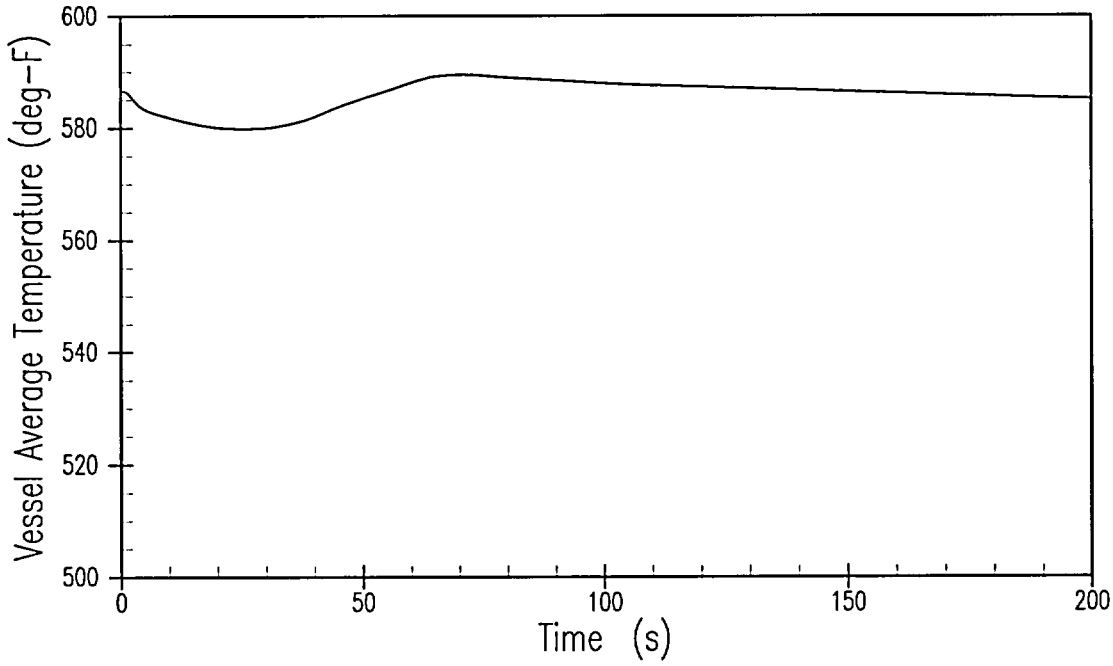


Figure 5.1.3-4 Representative Transient Response to Dropped RCCA – Vessel Average Temperature versus Time

5.1.4 Chemical and Volume Control System Malfunction (USAR Section 14.1.4)

Accident Description

Reactivity can be added to the core by feeding primary-grade water into the RCS via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation under strict administrative controls, with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the RCS. The chemical and volume control system is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the reactor makeup water control valve provides makeup water to the RCS that can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. For makeup water to be added to the RCS at pressure, the charging pumps must be running in addition to the reactor makeup water pumps.

The rate of addition of unborated makeup water to the RCS when it is not at pressure is limited by the capacity of the reactor makeup water pumps and the three charging pumps. Normally, two charging pumps are operated, one in manual and one in automatic control.

In order to dilute, two separate operations are required:

- The operator must switch from the automatic makeup mode to the dilute mode.
- The control switch must be activated.

Omitting either step prevents dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the chemical and volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

Method of Analysis

Boron dilutions during refueling, startup, and power operation are considered in this analysis (boron dilutions during hot standby, hot shutdown, and cold shutdown are not part of the KNPP licensing basis).

Dilution during Refueling

The maximum flow rate of unborated water that can be delivered to the RCS during refueling is assumed to be 120 gpm. This value assumes a single failure such that two charging pumps are delivering maximum flow.

A minimum RCS water volume of 1762.0 ft^3 is assumed, which is more conservative (that is, smaller) than the volume necessary to fill the reactor vessel up to the mid-plane of the nozzles plus the volume of one residual heat removal system (RHRS) train.

The ratio of the initial boron concentration to the maximum critical boron concentration during refueling is 1.34 (e.g., 2440 ppm / 1820 ppm). The boron concentration of the refueling water corresponding to a shutdown of at least 5% $\Delta k/k$ with all control rods in, is verified every reload cycle.

Dilution during Startup

In this mode, the plant is being taken from one long-term mode of operation, hot standby, to another, power. Typically, the plant is maintained in the startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation, rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. Conditions assumed for the analysis are:

- Dilution flow is the maximum capacity of two charging pumps, 120 gpm.
- A minimum RCS water volume of 5247.8 ft^3 corresponding to the active RCS volume (such as, not including the pressurizer volume) and accounts for 10-percent SGTP.
- The ratio of initial boron concentration to maximum critical boron concentration during startup is 1.125 (e.g., 1800 ppm / 1600 ppm). These are plant-specific values that are confirmed to be valid every cycle as part of the reload verification process.

This mode of operation is a transitory operational mode in which the operator intentionally dilutes and withdraws control rods to take the plant critical. During this mode, the plant is in manual rod control with the operator required to maintain a high awareness of the plant status. For a normal approach to criticality, the operator must manually initiate a limited dilution and subsequently manually withdraw the control rods. This process takes several hours. The Technical Specifications require that the operator determine the estimated critical position of the control rods prior to approaching criticality, thus assuring that the reactor does not go critical with the control rods below the insertion limits. Once critical, the power escalation must be sufficiently slow to allow the operator to manually block the source-range reactor trip. Failure to perform this manual action results in a reactor trip and an immediate shutdown of the reactor.

Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging pumps. A dilution flow rate of 120 gpm is assumed.

A minimum RCS water volume of 5247.8 ft^3 corresponding to the active RCS volume (such as, not including the pressurizer volume) accounts for 10-percent SGTP.

The ratio of initial boron concentration to maximum critical boron concentration during full power is 1.1125 (e.g., 1780 ppm / 1600 ppm). These are plant-specific values that are confirmed to be valid every cycle as part of the reload verification process.

With the reactor in automatic control, indication to the operator of the postulated dilution accident is provided by the rod insertion limits alarms (low and lo-lo setpoints) as the control rods are automatically inserted to compensate for the reactivity increase. The operator isolates the reactor makeup water source and initiates reboration.

If the reactor is in the manual control mode, the initial indication of a dilution accident is provided to the operator by using nuclear power and T_{avg}. Since the plant is under manual control, the operator is expected to follow these parameters closely and to react properly by further inserting the rods. In this fashion, manual control resembles the case with automatic control, with the operator taking the necessary steps to borate no later than when the manual insertion of the rods reaches the rod insertion limits.

If, however, the operator fails to take note of this slow change in reactivity, the following three alarms alert the operator to the dilution accident:

- T_{avg} - T_{ref} deviation alarm
- High flux, rod stop, and alarm
- OTΔT rod stop and turbine runback alarm

If the operator fails to take appropriate action on these alarms, the reactor trips on OTΔT. Dilution is indicated by constantly rising nuclear power and temperature and the absence of changes in rod position. Moreover, intermediate-range and source-range nuclear instrumentation system are available after the trip.

Once dilution has been identified, the operator terminates the flow of non-borated water. Following isolation of the reactor makeup water, the operator will reborate the RCS. The manner in which the boration is performed has no impact on the USAR analysis.

Operator Action Time Requirements

Analyses to determine the extent of fuel cladding damage and the overpressurization of the RCS are not done for this event. Instead, a calculation is performed to determine the amount of time available for operator action prior to the loss of the plant shutdown margin due to the dilution. Fifteen minutes for the at-power and startup conditions and thirty minutes for the refueling condition of plant operation from the initiation of the event are the criteria outlined in the earliest Standard Review Plan (SRP), Section 15.4.6 (dated September 1975). If these operator action times are met, it can be concluded that the fuel cladding damage and RCS overpressurization criteria are also satisfied.

Results

Dilution during Refueling

For dilution during refueling, the minimum time required for the shutdown margin to be lost and the reactor to become critical is 31.60 minutes.

For Dilution during Startup

For dilution during startup, the minimum time required for the shutdown margin to be lost and the reactor to become critical is 28.75 minutes.

For Dilution during Full-Power Operation

With the reactor in automatic control at full power, the power and temperature increase from boron dilution results in the insertion of the RCCAs and decrease in shutdown margin. Continuation of dilution and RCCA insertion would cause the assemblies to reach the minimum limit of the rod insertion monitor. Before reaching this point, however, two alarms are actuated to warn the operator of the accident condition. The first of these, the low insertion limit alarm, alerts the operator to initiate normal boration. The other, the lo-lo insertion limit alarm, alerts the operator to follow emergency boration procedures. The low alarm is set sufficiently above the lo-lo alarm to allow normal boration without the need for emergency procedures. If dilution continues after reaching the lo-lo alarm, it takes approximately 25.06 minutes before the total shutdown margin is lost due to dilution. Adequate time, therefore, is available following the alarms for the operator to determine the cause, isolate the reactor makeup water source, and initiate reboration.

With the reactor in manual control, if no operator action is taken, the power and temperature rise causes the reactor to reach the OTΔT trip setpoint. The boron dilution accident in this case is essentially identical to an RCCA withdrawal accident at power. Prior to the OTΔT trip, an OTΔT alarm and turbine runback would be actuated. There is time available (~22.68 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the reactor makeup water source, and initiate reboration before the reactor can return to criticality.

Conclusion

The time sequence of events is provided in Table 5.1.4-1. The boron dilution analyses at refueling, startup, and full-power conditions show the acceptability of the power uprating.

Table 5.1.4-1 Sequence of Events – Chemical and Volume Control System Malfunctions		
	Event	**Time (minutes)**
Refueling	Dilution begins	0
	Shutdown margin is lost	> 30
Startup	Dilution begins	0
	Shutdown margin is lost	> 15
At Power		
Automatic Reactor Control	Dilution begins	0
	Shutdown margin is lost	> 15
Manual Reactor Control	Dilution begins	0
	OTΔT reactor trip signal reached	2.38
	Rod motion begins	2.41
	Shutdown margin is lost (if dilution continues after trip)	> 17.38

5.1.5 Startup of an Inactive Reactor Coolant Loop (USAR Section 14.1.5)

If the plant were to operate with one RCP out of service, there would be reverse flow through the inactive loop due to the pressure difference across the reactor vessel and because there are no isolation valves or check valves in the reactor coolant loops. The cold-leg temperature in the inactive loop is identical to the cold-leg temperature of the active loop (the reactor core inlet temperature). If the reactor is operated at power with an inactive loop, and assuming that the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop. Therefore, with the reverse flow, the hot-leg temperature of the inactive loop would lower than the reactor core inlet temperature.

Starting the idle RCP without first bringing the hot-leg temperature of the inactive loop close to the core inlet temperature would result in an injection of cold water into the core. This injection of cold water into the core would cause a reactivity insertion, and subsequently a power increase due to the effects of moderator density reactivity feedback.

Sequence of Events and Systems Operation

Following the startup of an inactive RCP, flow in the inactive reactor coolant loop will accelerate to full flow in the forward direction over a period of several seconds. The KNPP Technical Specifications limit the reactor power to < 2-percent rated thermal power (RTP) when only one RCP is in operation. At this power level, the hot-leg temperature of the inactive loop would already be very close to the core inlet temperature.

The maximum initial core power level that needed to be considered for the startup of an inactive loop transient, therefore, was also low. Under these conditions, an analysis of this event was judged not to be necessary because the power level would be limited to a range that is non-limiting with respect to the minimum DNBR criterion. In this case, the resulting conditions following this event would be less limiting than normal steady-state full-power operation.

Conclusions

The startup of an inactive reactor coolant loop event results in an increase in reactor vessel flow while the reactor is maintained at a power level that is non-limiting with respect to minimum DNBR (less than 2 percent of nominal). No analysis is required to show that the DNBR limit is satisfied for this event.

5.1.6 Excessive Heat Removal Due to Feedwater System Malfunctions (USAR Section 14.1.6)

A change in steam generator feedwater conditions that results in an increase in feedwater flow or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. Such changes in feedwater flow or feedwater temperature are a result of a failure of a feedwater control valve or feedwater bypass valve, failure in the feedwater control system, or operator error.

The occurrence of these failures that result in an excessive heat removal from the plant primary coolant system cause the primary-side temperature and pressure to decrease significantly. The existence of a negative moderator and fuel temperature reactivity coefficients, and the actions initiated by the reactor rod control system can cause core reactivity to rise, as the primary-side temperature decreases. In the absence of the RPS reactor trip or other protective action, this increase in core power, coupled with the decrease in primary-side pressure, can challenge the core thermal limits.

Accident Description

Feedwater Temperature Reduction

An extreme example of excessive heat removal from the RCS is the transient associated with the accidental opening of the feedwater bypass valve, which diverts flow around the low-pressure feedwater heaters. The function of this valve is to maintain net positive suction head on the main feedwater pump in the event that the heater drain pump flow is lost; such as, following a large-load reduction. In the event of an accidental opening of the feedwater bypass valve, there is a sudden reduction in feedwater inlet temperature to the steam generators. This increased subcooling would create a greater load demand on the RCS due to the increased heat transfer in the steam generator.

With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator temperature coefficient. However, the rate of energy change is reduced as load and feedwater flow decrease, so that the transient is less severe than the full-power case.

The net effect on the RCS due to a reduction in feedwater temperature is similar to the effect of increasing secondary steam flow; that is, the reactor will reach a new equilibrium condition at a power level corresponding to the new steam generator ΔT. The overpower/overtemperature protection (high neutron flux, OTΔT, and OPΔT trips) prevent any power increase that could lead to a DNBR lower than the safety analysis limit value.

Feedwater Flow Increase

Another example of excessive heat removal from the RCS is a common-mode failure in the feedwater control system that leads to the accidental opening of the feedwater regulating valves (FW-7A and FW-7B) to both steam generators. Valves FW-7A and FW-7B could fail open due to a high output signal to the feedwater control system from any one of the following components:

- LM-463F, steam generator level auto programmer mode controller

- LM-463H, steam generator level program median selector
- LM-463D, current source for steam generator level minimum setpoint
- LM-463C, lead/lag circuit

This results in the valves stepping open 20 percent from their current position followed by a 20-percent step open every 5 minutes after that until full open. Accidental opening of the feedwater regulating valves results in an increase of feedwater flow to both steam generators, causing excessive heat removal from the RCS. At power, excess feedwater flow causes a greater load demand on the primary side due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-side temperature, and, therefore, a reactivity insertion due to the effects of the negative moderator temperature coefficient. The resultant decrease in the average temperature of the core causes an increase in core power due to moderator and control system feedback. This transient is attenuated by the thermal capacity of the primary and secondary sides. If the increase in reactor power is large enough, the primary RPS trip functions (such as high neutron flux, OTΔT, or OPΔT) will prevent any power increase that can lead to a DNBR less than the safety analysis limit value. The RPS trip functions may not actuate if the increase in power is not large enough.

Continuous addition of cold feedwater after a reactor trip is prevented since the reduction of RCS temperature, pressure, and pressurizer level leads to the actuation of safety injection on low pressurizer pressure. The safety injection signal trips the main feedwater pumps, closes the feedwater pump discharge valves, and closes the main feedwater control valves.

Method of Analysis

Feedwater Temperature Reduction

The reduction in feedwater temperature is determined by computing conditions at the feedwater pump inlet following the opening of the heater bypass valve. These feedwater conditions are then used to recalculate a heat balance through the high-pressure heaters. This heat balance gives the new feedwater conditions at the steam generator inlet. The following assumptions are made:

a. Initial power level of 1780 MWt

b. Low-pressure heater bypass valve opens, resulting in condensate flow splitting between the bypass line and the low pressure heaters; the flow through each path is proportional to the pressure drops

An evaluation method was applied that demonstrates the decreased enthalpy caused by the feedwater temperature reduction is bounded by an equivalent enthalpy reduction that results from an excessive load increase incident (Section 5.1.7). No explicit analysis is performed.

Feedwater Flow Increase

The feedwater malfunction analysis is performed to demonstrate that the DNB design basis is satisfied. This is accomplished by showing that the calculated minimum DNBR is greater than the safety analysis limit DNBR. The overall analysis process is described as follows.

The feedwater system malfunction transient is analyzed using the RETRAN code. The RETRAN computer code is a flexible, transient thermal-hydraulic digital computer code, that has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) for PWR licensing applications (Reference 5-5). The main features of the program include a point kinetics and one-dimensional kinetics model, one-dimensional homogeneous equilibrium mixture thermal-hydraulic model, control system models, and two-phase natural convection heat transfer correlations. The results from the RETRAN computer code are used to determine if the DNB safety analysis limits for the excessive heat removal due to feedwater malfunction event are met.

Feedwater system failures including the accidental opening of the feedwater regulating valves have the potential of allowing increased feedwater flow to each steam generator that will result in excessive heat removal from the RCS. Therefore, it is assumed that the feedwater control valves fail in the fully open position allowing the maximum feedwater flow to both steam generators. Cases with and without automatic rod control initiated at hot full-power (HFP) conditions were considered. Also addressed is the initiation of a feedwater malfunction event from a HZP condition.

The following assumptions are made for the analysis of the feedwater malfunction event involving the accidental opening of the feedwater regulating valves:

a. The plant is operating at full-power (and no-load conditions for the HZP case) conditions with the initial reactor power, pressure, and RCS average temperatures assumed to be at the nominal values.

b. Uncertainties in initial conditions are included in the DNBR limit calculated using the RTDP methodology (Reference 5-1), where applicable (full-power cases).

c. The feedwater temperature of 437.1°F for the full-power cases is consistent with normal plant conditions. The no-load feedwater temperature of 198.0°F is assumed in the zero-power case.

d. The excessive feedwater flow event assumes accidental opening of the feedwater control valves with the reactor at full power with automatic and manual rod control, and zero power while modeling post reactor trip conditions with minimum shutdown margin. The feedwater flow malfunction results in a step increase to 150 percent of the nominal full-power feedwater flow to both steam generators.

e. Maximum (end of life) reactivity feedback conditions with a minimum Doppler-only power defect is conservatively assumed.

f. The heat capacity of the RCS metal and steam generator shell are ignored, thereby maximizing the temperature reduction of the RCS coolant.

g. The feedwater flow resulting from a fully open control valve is terminated by the steam generator hi-hi water level signal that closes all main feedwater control and feedwater control-bypass valves, trips the main feedwater pumps, closes all feedwater pump discharge valves, and trips the turbine generator.

The RPS features, including power-range high neutron flux, OPΔT, and turbine trip on hi-hi steam generator water level, are available to provide mitigation of the feedwater system malfunction transient.

Results

Feedwater Temperature Reduction

The opening of a low-pressure heater bypass valve causes a reduction in feedwater temperature which increases the thermal load on the primary system. The reduction in feedwater temperature is less than 33°F, resulting in an increase in heat load on the primary system of less than 10 percent of full power. The reduction in feedwater temperature due to a 10-percent step load increase is greater than 33°F. The increased thermal load, due to the opening of the low-pressure heater bypass valve, thus results in a transient very similar, but of reduced magnitude, to the 10-percent step load increase incident described in Section 5.1.7. No transient results are presented, as no explicit analysis is performed.

Feedwater Flow Increase

The results of the analyses demonstrate that both the HFP cases and zero-power case meet the applicable DNBR acceptance criterion.

The most limiting case is the excessive feedwater flow from a full-power initial condition with automatic rod control. This case gives the largest reactivity feedback and results in the greatest power increase. A turbine trip, which results in a reactor trip, is actuated when the steam generator water level in either steam generator reaches the hi-hi water level setpoint. Assuming the reactor to be in manual rod control results in a slightly less severe transient. The rod control system is not required to function for this event. However, assuming that the rod control system is operable yields a slightly more limiting transient.

The excessive feedwater flow from a zero-power condition models a HZP post-trip condition (that is, HZP stuck rod coefficients, minimum shutdown margin) with maximum reactivity feedback conditions (end of life). The limiting HZP feedwater malfunction conditions were analyzed and confirmed that the calculated minimum DNBR is above the safety analysis DNBR limit. Therefore, the applicable DNBR acceptance criterion is met.

For each excessive feedwater flow case, continuous addition of cold feedwater is prevented by automatic closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on hi-hi steam generator water level. In addition, the feedwater discharge isolation valves will automatically close upon receipt of the feedwater pump trip signal.

Following turbine trip, the reactor will automatically be tripped, either directly due to the turbine trip or due to one of the reactor trip signals discussed in Section 5.1.9 (Loss of External Electrical Load). If the reactor was in automatic rod control, the control rods would be inserted at the maximum rate following the turbine trip, and the resulting transient would not be limiting in terms of peak RCS or MSS pressure.

Table 5.1.6-1 shows the time sequence of events for the HFP feedwater malfunction transients analyzed at full-power initial conditions assuming manual and automatic rod control. Table 5.1.6-2 shows the time sequence of events for the HZP feedwater malfunction transient. Figures 5.1.6-1 through 5.1.6-5 show

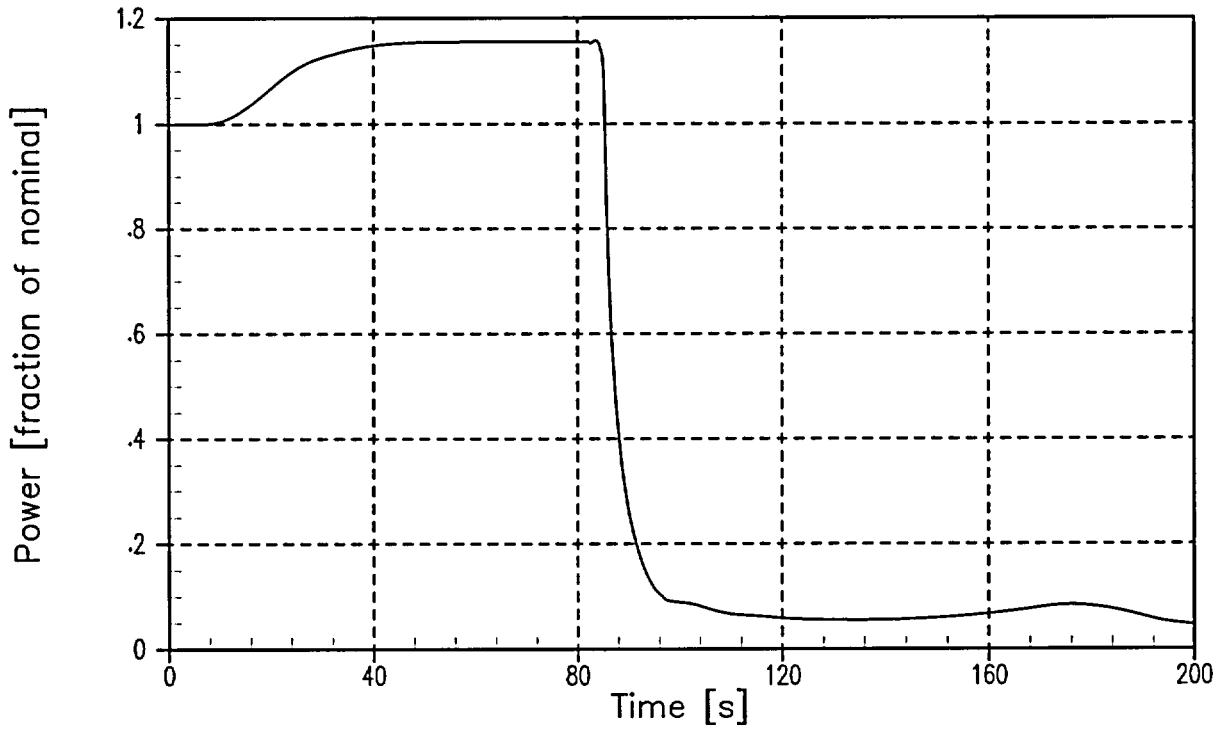
transient responses for various system parameters during a feedwater system malfunction initiated from HFP conditions without automatic rod control (manual control). Figures 5.1.6-6 through 5.1.6-10 show transient responses for various system parameters during a feedwater system malfunction initiated from HFP conditions with automatic rod control.

Conclusions

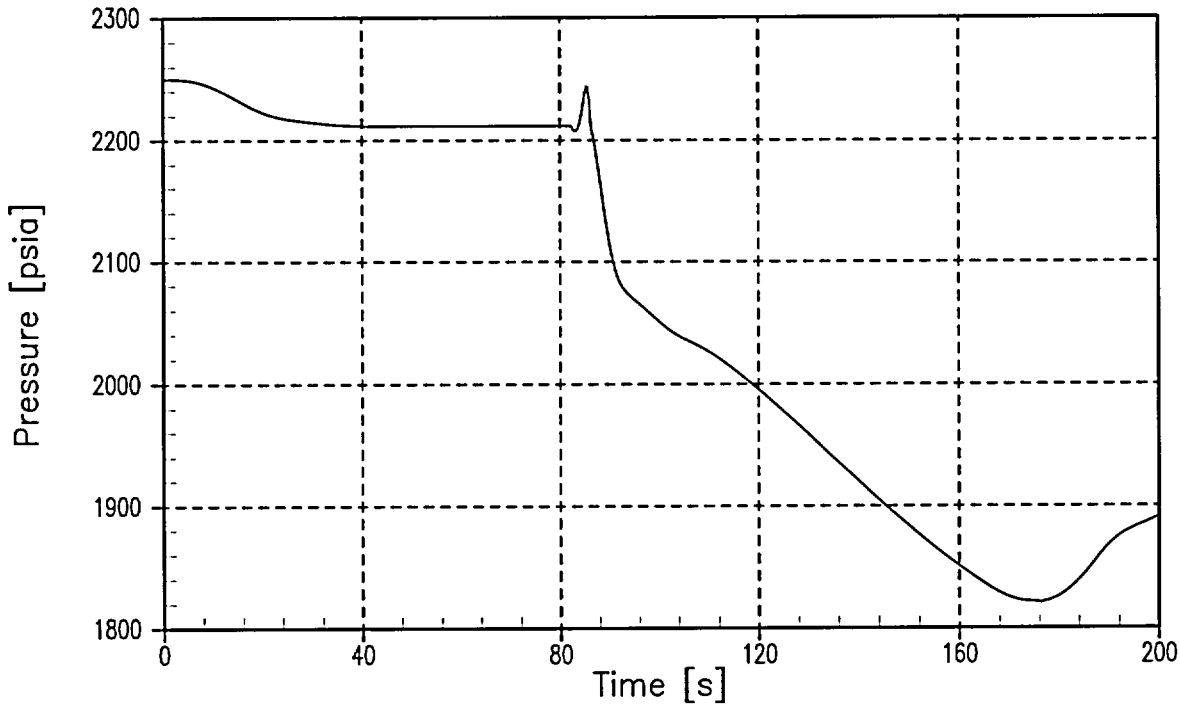
Feedwater system malfunction transients involving a reduction in feedwater temperature or an increase in feedwater flow rate have been analyzed or evaluated. These transients show an increase in reactor power due to the excessive heat removal in the steam generators. With respect to the feedwater temperature reduction transient (accidental opening of the feedwater bypass valve), it was determined to be less severe than the excessive load increase incident (see USAR Section 14.1.7); no explicit analysis is performed. Based on results presented in Section 5.1.7, the applicable acceptance criteria for the feedwater temperature reduction transient have been met. Analyses of the accidental opening of the feedwater regulating valves were performed from a full-power initial condition with and without automatic rod control, and from a zero-power initial condition. It has been demonstrated that considerable margin to the safety analysis acceptance criteria exists throughout the transient. Therefore, the DNB design basis is satisfied. Hence, no fuel damage is predicted.

Table 5.1.6-1 Sequence of Events for Feedwater System Malfunction Event at Full Power		
	Time (Seconds)	
Event	Without Automatic Rod Control	With Automatic Rod Control
Main Feedwater Control Valves Fail Full Open	0.0	0.0
Hi-Hi Steam Generator Water Level Trip Setpoint is Reached	80.8	81.5
Reactor Trip Occurs Due to Turbine Trip	83.7	84.4
Turbine Trip Occurs Due to Hi-Hi Steam Generator Level	81.9	82.6
Minimum DNBR Occurs	83.4	84.1
Feedwater Isolation Valves Fully Closed	166.0	166.7
Results		
Peak Nuclear Power, Fraction of Initial	1.164	1.169
Peak Core Heat Flux, Fraction of Initial	1.157	1.162
Minimum DNBR	1.730	1.709
Safety Analysis Limit DNBR (WRB-1 correlation limit)	1.34	1.34

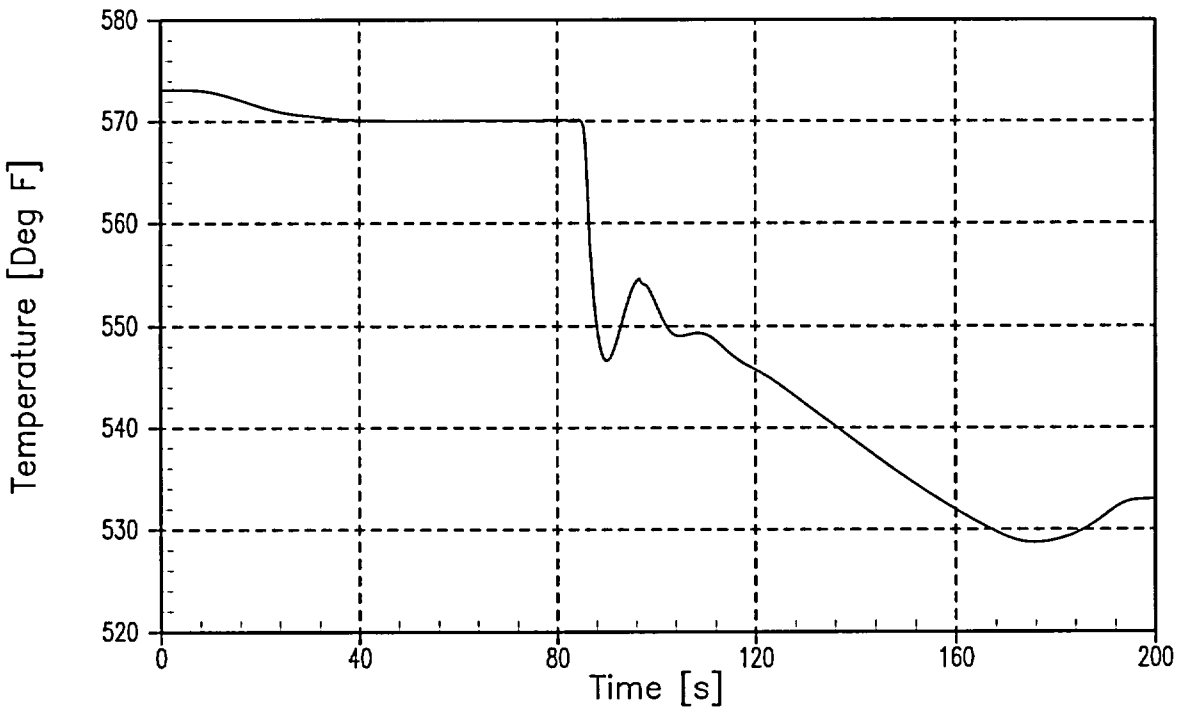
Table 5.1.6-2 Sequence of Events for Feedwater System Malfunction Event at Zero Power	
Event	**Time (Seconds)**
Main Feedwater Control Valves Fail Full Open	0.0
Hi-Hi Steam Generator Water Level Trip Setpoint is Reached	51.3
Feedwater Isolation Valves Fully Closed	125.4
Results	
Peak Nuclear Power, Fraction of Initial	0.207
Peak Core Heat Flux, Fraction of Initial	0.210
Minimum DNBR	2.837
Safety Analysis Limit DNBR (W-3 correlation limit)	1.472



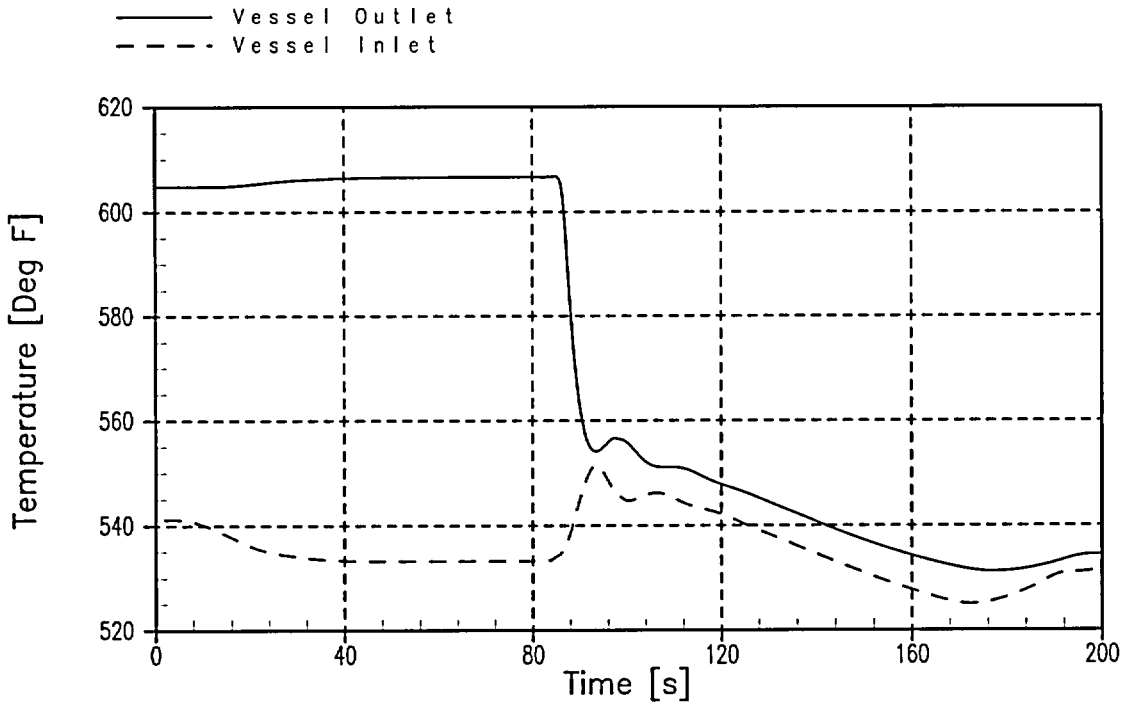
Figures 5.1.6-1 Feedwater Malfunction Event at Full Power with Manual Rod Control – Reactor Power versus Time



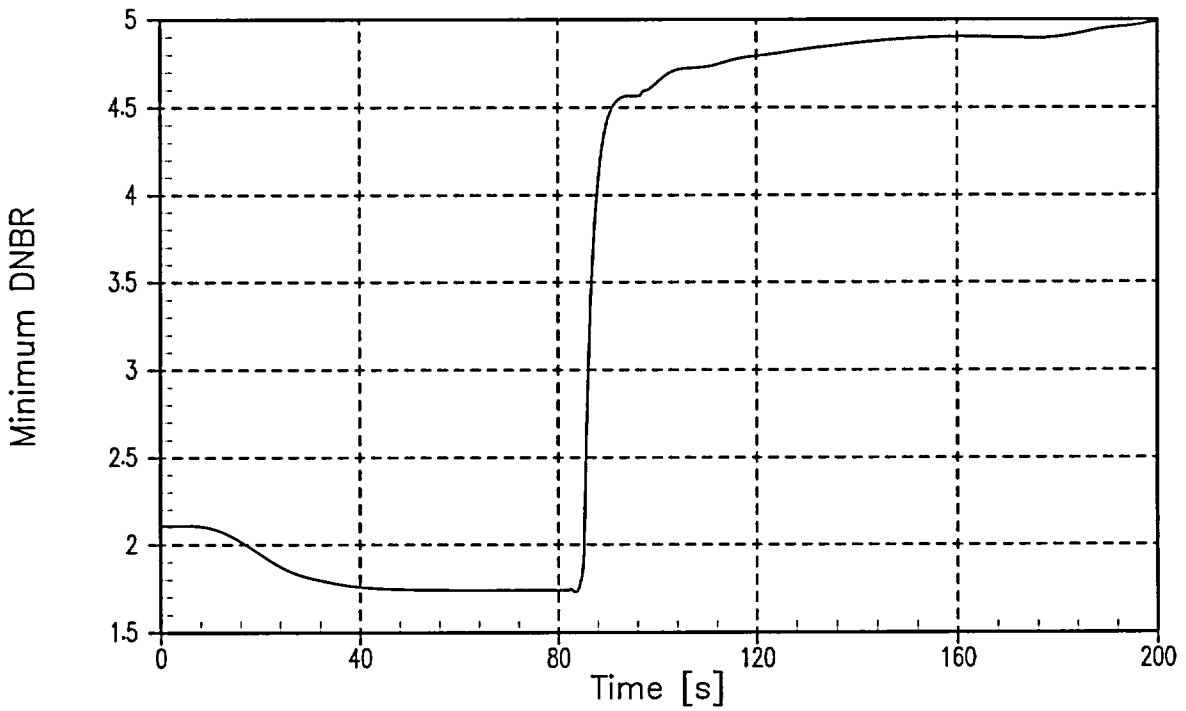
Figures 5.1.6-2 Feedwater Malfunction Event at Full Power with Manual Rod Control – Pressurizer Pressure versus Time



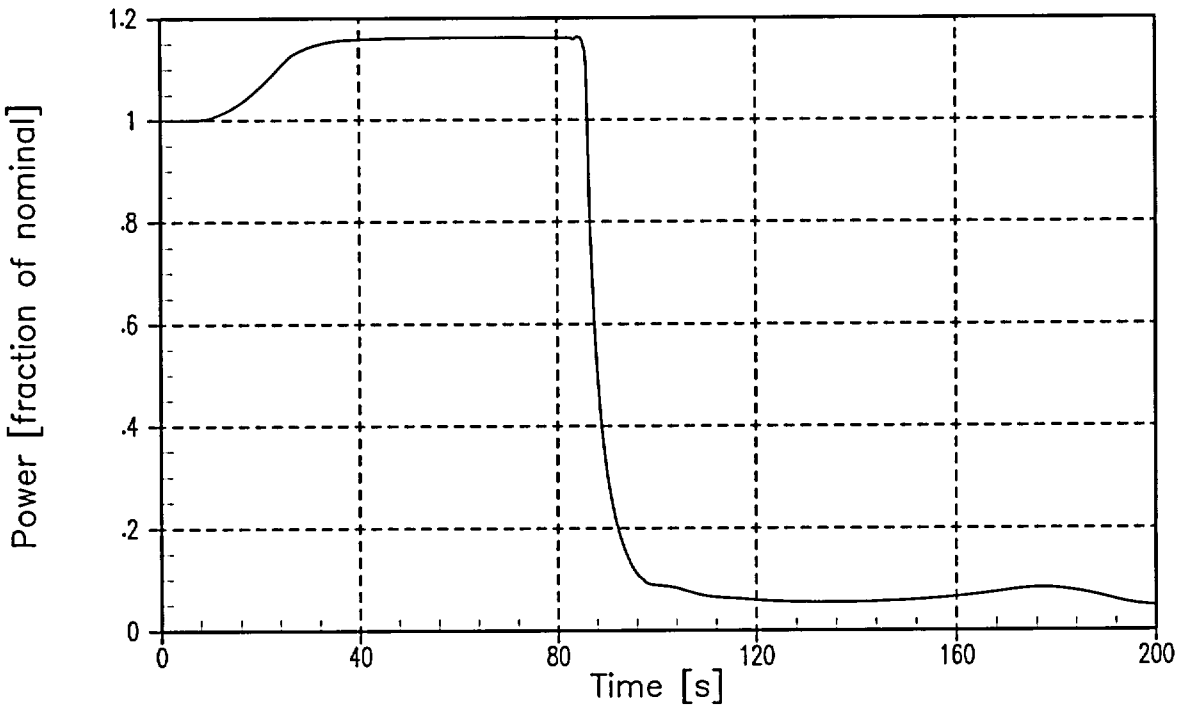
Figures 5.1.6-3 Feedwater Malfunction Event at Full Power with Manual Rod Control – Core Average Temperature versus Time



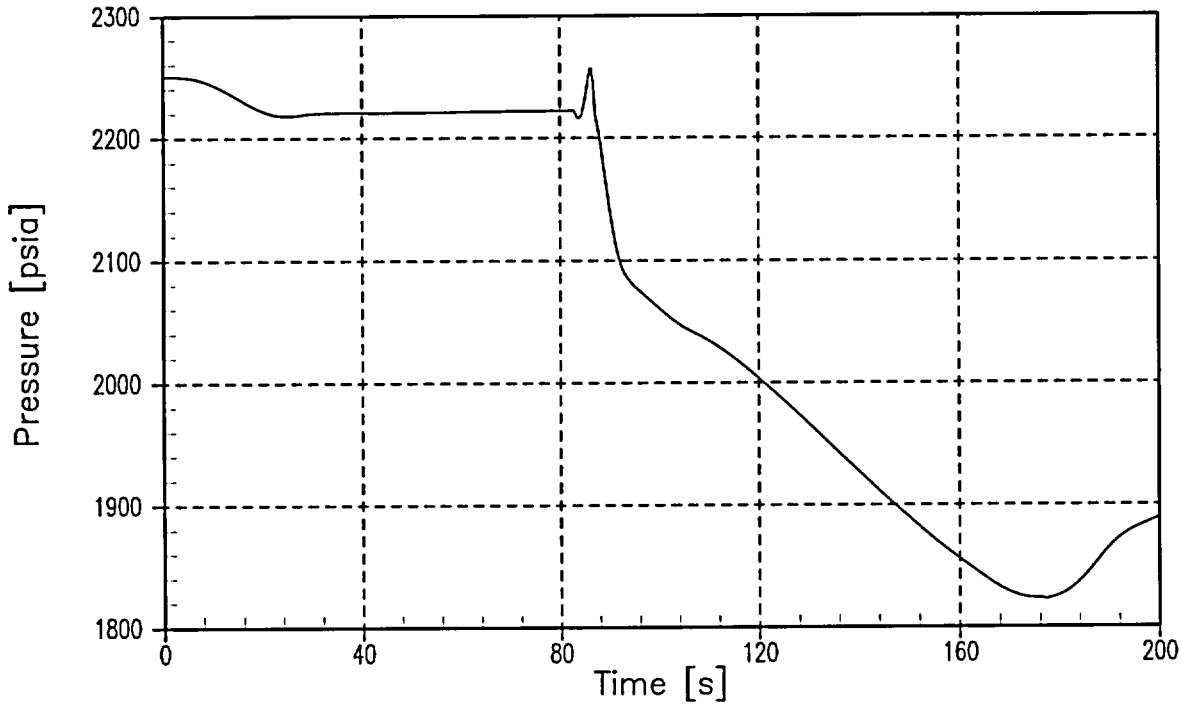
Figures 5.1.6-4 Feedwater Malfunction Event at Full Power with Manual Rod Control – Vessel Outlet and Inlet Temperatures versus Time



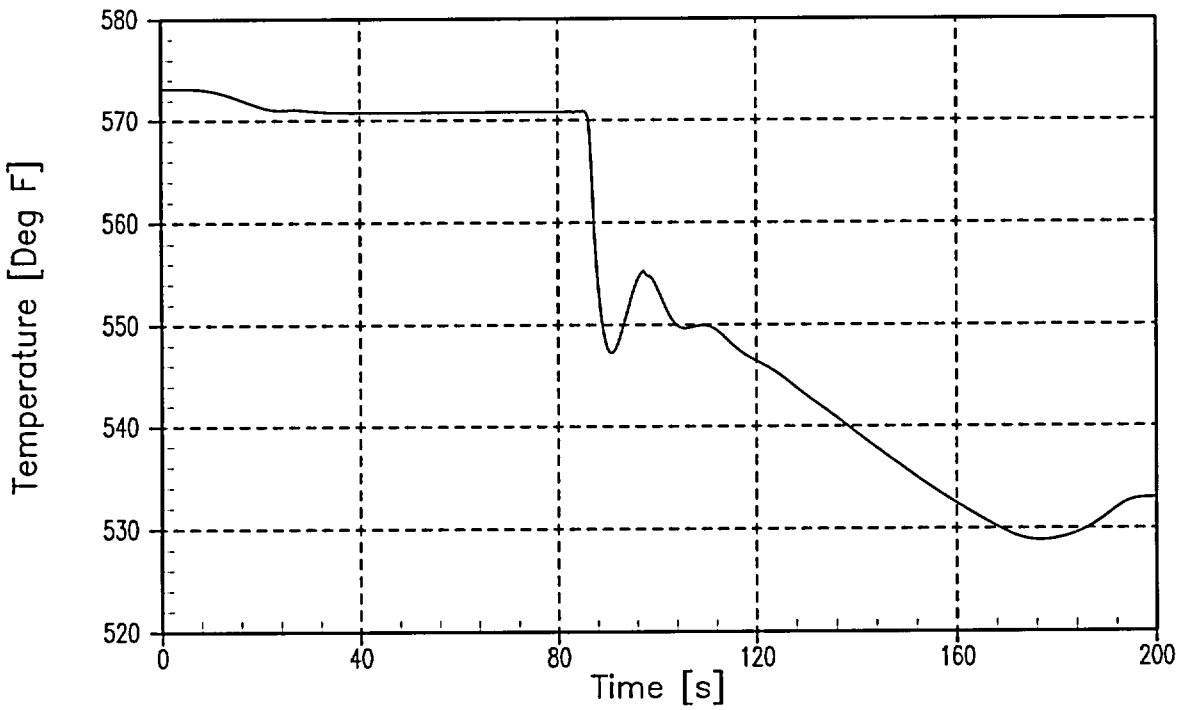
Figures 5.1.6-5 Feedwater Malfunction Event at Full Power with Manual Rod Control – Minimum DNBR versus Time



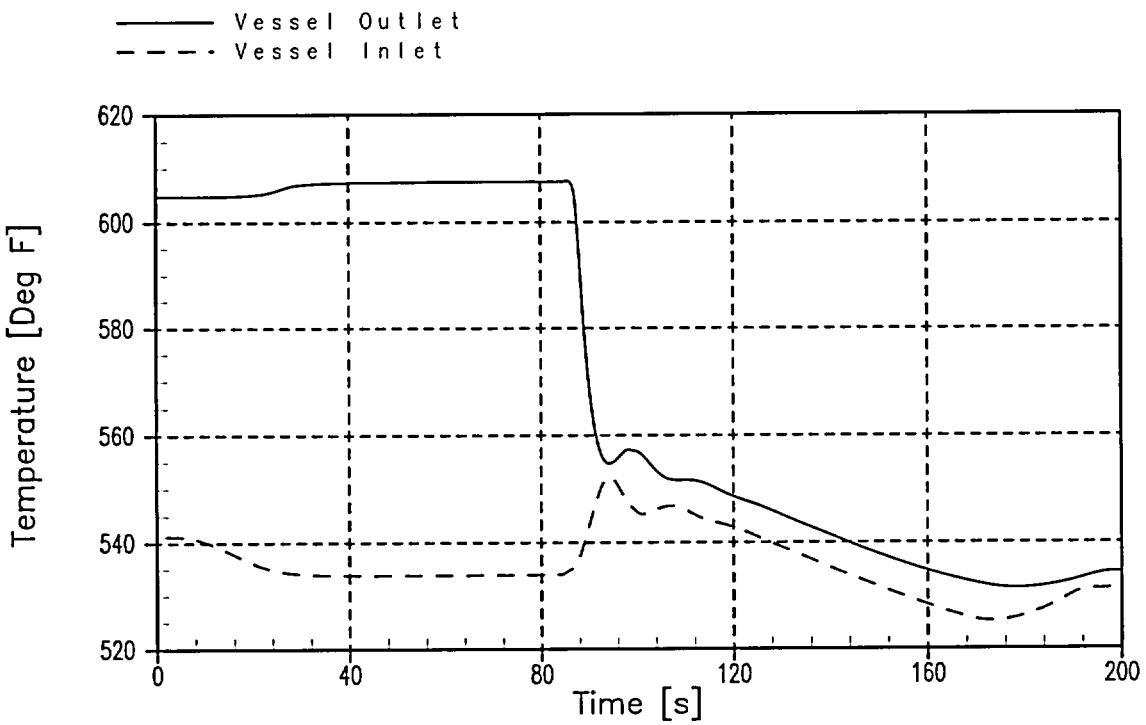
Figures 5.1.6-6 Feedwater Malfunction Event at Full Power with Automatic Rod Control – Reactor Power versus Time



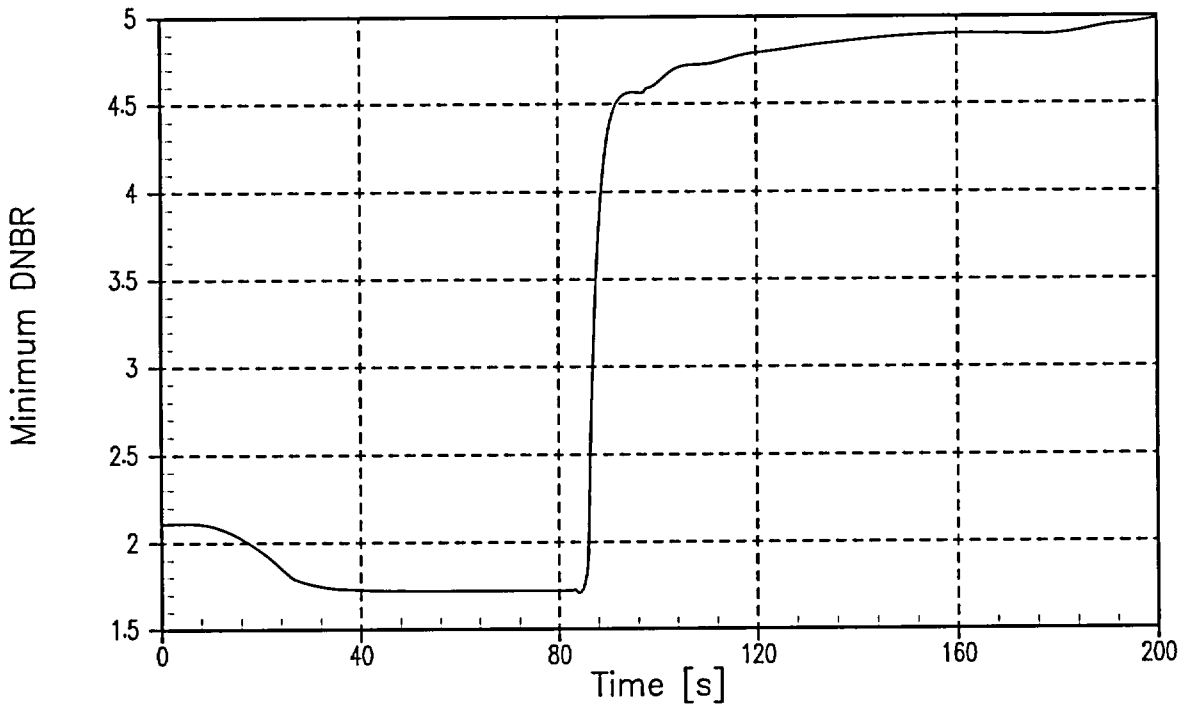
Figures 5.1.6-7 Feedwater Malfunction Event at Full Power with Automatic Rod Control – Pressurizer Pressure versus Time



Figures 5.1.6-8 Feedwater Malfunction Event at Full Power with Automatic Rod Control – Core Average Temperature versus Time



Figures 5.1.6-9 Feedwater Malfunction Event at Full Power with Automatic Rod Control – Vessel Outlet and Inlet Temperatures versus Time



Figures 5.1.6-10 Feedwater Malfunction Event at Full Power with Automatic Rod Control – Minimum DNBR versus Time

5.1.7 Excessive Load Increase Incident (USAR Section 14.1.7)

Accident Description

An excessive load increase incident is defined as an event resulting in a rapid increase in the steam generator steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10-percent step load increase or a 5-percent per minute ramp load increase (without a reactor trip) in the range of 15 to 95 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the RPS.

This accident could result from either an administrative violation, such as excessive loading by the operator, or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals; that is, a high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or turbine trip has occurred.

The possible consequence of this accident (assuming no protective functions) is a DNB with subsequent fuel damage. Note that the accident is typically characterized by an approach of parameter values to the protection setpoints without the setpoints actually being reached.

Method of Analysis

The excessive load increase incident is analyzed to show that:

- The integrity of the core is maintained typically without the RPS being actuated (that is, the minimum DNBR remains above the safety analysis limit value).
- The peak RCS and MSS pressures remain below 110 percent of the design values.
- The pressurizer does not become water-solid.

Of these, the primary concerns are DNB and ensuring that the DNBR limit is met.

However, as discussed earlier, this transient does not typically result in the actuation of any RPS function (that is, no reactor trip). The effect of this transient on the minimum DNBR was evaluated by applying conservatively large deviations on the initial conditions for power, average coolant temperature, and pressurizer pressure at the normal full-power operating conditions in order to generate a limiting set of statepoints. These deviations bound the variations that could occur as a result of an excessive load increase incident and are only applied in the direction that had the most adverse impact on DNBR (increased power and coolant temperature, and decreased pressure). The reactor condition statepoints (power, temperature, and pressure) were then compared to the conditions corresponding to operation at the DNB safety analysis limit (shown previously in Figure 5.1-1).

The results of the evaluation performed to support the KNPP FU/PU Program showed that the minimum DNBR would remain above the safety analysis limit value. Therefore, it was determined that a more detailed analysis using the RETRAN code was not necessary for implementation of the FU/PU Program. The USAR will be updated to state that a representative excessive load increase transient is presented as opposed to one specific to the FU/PU Program.

Conclusions

In the event of an excessive load increase incident, (that is, a 10-percent step load increase), the minimum DNBR remains above the safety analysis limit value, thereby precluding fuel or cladding damage. Peak RCS and MSS pressures do not challenge applicable pressure limits.

5.1.8 Loss of Reactor Coolant Flow (USAR Section 14.1.8)

The loss of reactor coolant flow events are categorized as follows in the KNPP USAR:

- Flow coastdown accidents
- Locked-rotor accident

The first category includes the partial and complete loss of reactor coolant flow, and the reactor coolant pump underfrequency events. The second category includes the hypothetical event that addresses an instantaneous seizure of an RCP rotor.

Partial Loss of Reactor Coolant Flow

Accident Description

The partial loss-of-coolant-flow accident can result from a mechanical or electrical failure in an RCP, or from a fault in the power supply to the RCP. If the reactor is at power at the time of the accident, the immediate effect of loss-of-coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

Normal power for the pumps is supplied through individual buses connected to the generator and the offsite power system. When a generator trip occurs, the buses continue to be supplied from external power lines, and the pumps continue to supply coolant to the core.

The necessary protection against a partial loss-of-coolant-flow accident is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two-out-of-three low flow signals. Above 10-percent nuclear instrumentation system (NIS) power (Permissive 8), low flow in either loop will actuate a reactor trip. Above 10-percent NIS and 10-percent turbine power (Permissive 7), low flow in both loops will actuate a reactor trip.

Method of Analysis

The loss of an RCP with both loops in operation event is analyzed to show that: (1) the integrity of the core is maintained as the DNBR remains above the safety analysis limit value, and (2) the peak RCS and

secondary system pressures remain below the design limits. Of these, the primary concerns are DNB and assuring that the DNBR limit is met.

The loss of an RCP event is analyzed with two computer codes. First, the RETRAN computer code is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary-system pressure and temperature transients. The VIPRE computer code is then used to calculate the hot-channel heat flux transient and DNBR, based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from RETRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

This event is analyzed with the RTDP. Initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values. Minimum measured flow is also assumed. A conservatively large absolute value of the Doppler-only power coefficient is used, along with the most-positive MTC limit for full-power operation (0 pcm/°F). These assumptions maximize the core power during the initial part of the transient when the minimum DNBR is reached.

A limiting EOC DNB axial power shape is assumed in VIPRE for the calculation of DNBR. This shape provides the most limiting minimum DNBR for the loss-of-flow events.

A conservatively low trip reactivity value (3.5-percent $\Delta\rho$) is used to minimize the effect of rod insertion following reactor trip and maximize the heat flux statepoint used in the DNBR evaluation for this event. This value is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. A conservative trip reactivity worth versus rod position was modeled in addition to a conservative rod drop time (1.8 seconds to dashpot). The trip reactivity versus rod position curve is confirmed to be valid as part of the Reload Safety Analysis Checklist (RSAC) verification process.

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics. Also, it is based on conservative estimates of system pressure losses.

A maximum, uniform, SGTP level of 10 percent was assumed in the RETRAN analysis. However, a core flow reduction of 1.1 percent, which addresses the potential reactor coolant flow asymmetry associated with a maximum loop-to-loop SGTP imbalance of 10 percent, was applied.

Results

Figures 5.1.8-1 through 5.1.8-8 illustrate the transient response for the loss of an RCP with both loops in operation. The minimum DNBR is 1.646/1.666 (thimble/typical), which occurred at 3.5 seconds (DNBR limit: 1.34/1.34 (thimble/typical)).

The calculated sequence of events table is shown in Table 5.1.8-1. This transient trips on a low primary reactor coolant flow trip setpoint, which is assumed to be 86.5 percent of loop flow. Following reactor trip, the affected RCP will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the remaining pump still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.