ATTACHMENT 1

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 187

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Consideration

Introduction

Nuclear Management Company, LLC (NMC) intends to operate its Kewaunee Nuclear Power Plant (KNPP) reactor using Westinghouse 422 VANTAGE+ nuclear fuel with PERFORMANCE+ features (referred to subsequently as 422V+), commencing with the Cycle 26 core refueling presently scheduled for April of 2003.

The 422V+ fuel design proposed by this request is essentially the same as the fuel that the Nuclear Regulatory Commission (NRC) approved (Reference 1) for use at Point Beach Nuclear Plant (PBNP) as requested by Wisconsin Electric (References 2 and 3). Design features of the 422V+ fuel were described in detail by a Westinghouse topical report, WCAP-12610-P-A (Reference 4), which was referenced by the PBNP request. The NRC subsequently approved the PBNP request and issued amendments 193 and 198 for facility operating licenses DPR-24 and DPR-27, respectively.

Prior to the use of 422V+ fuel for full reloads, it is necessary to make changes to KNPP Technical Specifications (TS). Those changes are shown in attachments to this request. NMC requests NRC permission for the TS amendments and provides associated USAR changes as information for NRC use in evaluating this request.

NMC worked with Westinghouse to review the KNPP design and evaluate use of 422V+ fuel. The reanalysis was performed using NRC approved, Westinghouse computer codes and methods, thereby establishing a new Record of Analysis which is documented in the Technical Design Basis for the change (Attachment 4) and in the USAR markups (Attachment 5). The analyses were performed assuming a KNPP reactor power level of 1772 MWt rather than the currently licensed level of 1650 MWt. The use of the higher power level in the technical design basis calculations adds conservatism to analytical results; however, operation at the higher level is not requested by this proposal.

The Westinghouse Report serves as a reference safety evaluation and analysis report for the region-by-region reload transition from the KNPP Cycle 25 reactor core to subsequent cores containing a mix of 422V+ and earlier designs, and ultimately to a homogeneous core loaded solely with 422V+ fuel. Thus, it will be a reference for future KNPP Reload Safety Evaluations (RSE). Each RSE will be conducted to show cycle-specific conformance within the parameters established by the RTSR or require cycle-specific analyses or evaluations to ensure safe operation of each core design, including Cycle 26.

NRC approval of this TS amendment is requested by February 28, 2003, for implementation during the Cycle 26 refueling outage.

Description of Changes to KNPP Technical Specifications (TS)

The following is a list of changes to the Technical Specification, a description of each change, and the reason that the change is being made:

TS Section	Description of Changes	Reason
1.p	Changed to Dose Equivalent I-131 definition and dose conversion factors	Revised methodology and assumptions that are derived from TID-14844 to ICRP-30 consistent with radiological consequences analyses.
2.1.b Basis TS 2.1	Added WRB-1 Correlation	Change reflects the new DNB correlation used in analyses for Westinghouse 422 V+ fuel.
Basis TS 2.2	Modified PORVs basis	Clarified the design basis for the Pressurizer PORVs by indicating the design basis events and the pressure value acceptance criterion which is applicable to the PORVs.
2.3.a.3.A	Changed Overtemperature ΔT term descriptions	Changed to be consistent with analysis assumptions.
2.3.a.3.B	Changed Overpower ΔT term descriptions	Changed to be consistent with analysis assumptions. There is no f(ΔI) penalty for this function.
3.1.c Basis TS 3.1.c Basis TS 3.4.c	Changed specific activity values, replaced 10 CFR 100 with 10 CFR 50.67, and deleted Figure 3.1-3, the Dose Equivalent I-131 figure.	Changes made for consistency with ICRP-30 and radiological consequence analyses.
Basis TS 3.1.a	Changed DNBR value	Change is associated with 422 V+ fuel.
3.4.d	Changed the units "μCi/cc" to "μCi/gram	Changes made for consistency with ICRP-30 and radiological consequence analyses.
Basis TS 3.4	Changed Maximum full-power steam flow and AFW operable description.	Change made to be consistent with analyses.
3.10.b 3.10.b.7 Basis TS 3.10.b Basis TS 3.10.b	Changed the actions for $F_Q^N(Z)$, $F_{\Delta H}^N$, and $F_Q^{EQ}(Z)$	These actions were changed to be consistent with the FQ surveillance methodology and 422 V+ fuel change.
3.10.b.7 through 3.10.b.13 Basis TS 3.10.b	Changed the surveillance and actions for axial flux difference	These changes were changed to be consistent with the relaxed axial offset control (RAOC) methodology and 422 V+ fuel change.
3.10.m Basis TS 3.10.k Basis TS 3.10.l Basis TS 3.10.m	Changed the surveillances for reactor coolant flow and bases for pressure, temperature and flow.	These changes are made to be consistent with analysis assumptions.
6.9	Added reference	Reference is added for COLR methods documentation.

TS Section	Description of Changes	Reason
2.6	Added new $F_Q^N(Z)$, $F_{\Delta H}^N$, and $F_Q^{EQ}(Z)$ limits for 422 V+ fuel	Maximum F_Q and $F_{\Delta H}^N$ are generated that yield acceptable results based upon the safety analysis limits. For F_Q see Section 3.5. For $F_{\Delta H}^N$ see Sections 3.6 and 4.2.
2.6	Replaced the V(z) function with W(z) function.	The W(z) function is used for consistency with the F_Q Surveillance Methodology.
2.7	Increased the part power multiplier for $F_{\Delta H}^N$.	This change was made for consistency with analysis inputs.
2.9	Changed the OTΔT function.	This change was made to reflect the impact of the transition to 422 V+ fuel and for consistency to the Non-LOCA Analysis.
2.10	Changed the OPΔT function.	This change was made to reflect the impact of the transition to 422 V+ fuel and for consistency to the Non-LOCA Analysis.
2.11	Revised the DNB Parameters (flow, pressure and temperature)	The flow change is made to be consistent with the Westinghouse Methodology for RCS flow measurement. The pressure and temperature change is made to reflect KNPP specific uncertainties, procedures, and calibration practices in generating the indicated values.
2.12	Revised Refueling Boron Concentration Value from 2200 ppm to 2250 ppm	Boron concentration change was made for consistency with the accident analyses.
Figure 1	Revised the Core Safety Limits Curve	This change was made to reflect the impact of the transition to 422 V+ fuel and for consistency to the Non-LOCA Analysis.
Figure 2	Revised Required Shutdown Margin (SDM)	The SDM change was made to provide additional fuel management flexibility and is consistent with accident analysis.
Figure 3	Revised Hot Channel Factor Normalized Operating Envelope K(z)	The K(z) curve was changed for consistency with the LOCA analysis.
Figure 5	Replaced the V(z) figure with the W(z) figures.	The W(z) function is used for consistency with the FQ Surveillance Methodology.

Safety Evaluation for Proposed Change to KNPP TS

Changes proposed by this request modify the KNPP TS to allow use of full reloads of Westinghouse 422V+ fuel beginning with Cycle 26. Discharged fuel will be replaced by 422V+ fuel during this and subsequent refueling operations, until the core contains only 422V+ fuel.

Compared with current Framatome/ANP design, 422V+ fuel has a slightly smaller fuel rod diameter. Other minor differences exist and are presented in Table 2-1 of the Technical Design Basis for the Transition to 422V+ Fuel in Attachment 4 (heretofore referred as Westinghouse Report). While the current fuel uses a zirconium alloy known as Zircaloy for fuel clad and for structural elements of the fuel assembly, 422V+ fuel uses a substantially similar zirconium alloy marketed under the Westinghouse ZIRLO trademark. This new name requires changes to TS and USAR references that do not presently cite ZIRLO as an acceptable material. Westinghouse 422V+ fuel with ZIRLO is mechanically, hydraulically, and chemically compatible with the KNPP reactor design and the Framatome/ANP fuel currently loaded in its core.

During fuel transition cycles, KNPP cores will contain combinations of partially burned Framatome/ANP fuel and 422V+ fuel, and fresh 422V+ fuel. The Westinghouse Report discusses fuel design changes and describes the safety analyses performed to bound core conditions containing any combination of these fuel designs, including a full core load of 422V+ fuel. The effect of 422V+ on new-fuel and spent-fuel handling and storage was analyzed and remains acceptable. For Cycle 26 and subsequent core loads, cycle-specific RSE will be performed using Westinghouse standard reload methodology (Reference 6) to verify that safety analaysis results satisfy their respective acceptance criteria.

Note that certain of the analyses performed for this request assumed a reactor power of 1772 MWt. No power uprate request is made at this time. Conclusions reached using the 1772 MWt assumption bound results using the current power level of 1650 MWt. Reactor trip set-points calculated using the higher power level also bound safe operation at the currently licensed power level.

Westinghouse reanalyzed many of the existing safety analyses as documented in Attachment 4, to revise or confirm assumptions and inputs to ensure their accuracy and that of the KNPP licensing basis. References 1-17, 1-18, and 1-19 document the final assumptions and input parameters used in the analyses. However, the individual calculation notes (Reference 5, Table 5.1-8) should be used as the final verification of all assumptions and parameters.

The Westinghouse Report documents extensive analyses that led to the findings summarized below. Together, these findings conclude that plant operation using 422V+ fuel is safe, either in combination with various mixes of existing fuel of other designs, or as the sole core-resident fuel.

- 422V+ fuel is mechanically compatible with current Framatome/ANP fuel assembly design, control rods, flux detectors inserted in instrumentation tubes, fuel handling equipment, and reactor internals (Chapters 2 and 6).
- Evaluation of 422V+ fuel for postulated faulted condition accidents, including LOCA and seismic events, demonstrate that the fuel assembly is structurally adequate to prevent grid-crush by resultant component stresses and grid impact forces (Chapters 2, 5, and 6).

- NMC and Westinghouse core design management using Reload Safety Evaluation (RSE) analyses will ensure that changes in nuclear characteristics of 422V+ fuel during the transition period and subsequent cycles will remain within the ranges analyzed in the Westinghouse Report. Alternatively, additional analyses or evaluations will be performed to demonstrate that the plant will be operated in a manner compliant with all safety criteria. (Chapter 3).
- 422V+ fuel is hydraulically compatible with current Framatome/ANP fuel (Chapters 2 and 4).
- Core design and safety analyses of 422V+ fuel confirm that parameters and conditions at an uprated 1772 MWt power level remain within licensed safety margins (Chapters 3, 4, 5 and 6).
- The analyses and evaluations summarized in Attachment 4 provide the design basis for Westinghouse RSE of future reloads using 422V+ fuel. (Chapters 2, 3, 4, 5, and 6).

The NRC has generically approved use of Westinghouse 422V+ fuel (Reference 4), and has generically approved (Reference 7) the use of a limited number of assemblies as lead test assemblies at KNPP. During Cycle 25 (Reference 8), NMC has confirmed that the fuel is compatible with the KNPP reactor and the existing Framatome/ANP fuel. Analysis of transition and equilibrium 422+ fuel in either mixed or homogenous core configurations remains within design basis limitations and safety margins. Thus, operation with 422V+ fuel remains bounded by design basis accident and transient analyses.

Commitment

Upon NRC approval of WCAP 12488, Addendum 2, "Revision to Design Criteria," confirmation of meeting the criterion will be submitted.

Significant Hazards Determination for Proposed Change to TS

NMC reviewed the proposed change in accordance with provisions of 10 CFR 50.92 and determined that it produces no significant hazards. The proposed change does not:

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

   The NRC generically approved Westinghouse 422V+ fuel assemblies for use in reactors substantially similar to KNPP. NMC used 422V+ fuel in the Lead-Test-Assembly Program during Cycle 25, as permitted by existing TS. 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. Reanalysis of postulated KNPP design basis accidents shows that reactor operation with 422V+ fuel remains within design basis limitations and safety margins. All design basis accidents and transients affected by the fuel upgrade were analyzed, and the results documented in the Westinghouse Report provided with this request. These analyses and evaluations show that use of 422V+ fuel is acceptable. The margin to safety is not exceeded in any instance. Pending approval of Addendum 2 to WCAP 12488 revising the current transient stress strain criteria, all design basis acceptance criteria will be satisfied. Changes to the technical specification that remain within the limits of the bounding accident analyses cannot change the probability or consequence of an accident previously evaluated. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

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

   Use of the 422V+ fuel is consistent with current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. The operational characteristics of 422V+ fuel are bounded by the safety analyses (Attachment 4). The 422V+ fuel design performs within existing fuel design limits. Thus, this proposal does not create the possibility of a new or different kind of accident.

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

   The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Set-points, or Limiting Conditions for Operation are determined. Licensed safety margins are maintained. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Analyses of design basis accidents and transients were performed using a power level greater than that currently licensed, thus rendering more conservative results than required. All safety analysis acceptance criteria are satisfied at this value and all KNPP safety requirements continue to be met. Use of 422V+ fuel as proposed by this amendment request is bounded by these analyses. Thus, changes proposed by this request do not involve a significant reduction in the margin of safety.

Environmental Considerations

This proposed amendment involves a change to the Technical Specifications. It does not modify any facility components located within the restricted area, as defined in 10 CFR 20. NMC has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. This proposed amendment accordingly meets the eligibility criteria for categorical exclusion set forth by 10 CFR 51.22(c)(9). Pursuant thereto, no environmental impact statement or environmental assessment need be prepared in connection with this amendment proposal.

References:

1. USNRC Safety Evaluation Report, "SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO 193 TO FACILITY OPERATING LICENSE NO. DPR-24 AND AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-27 WISCONSIN ELECTRIC POWER COMPANY POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 DOCKET NOS. 50-266 AND 50-301"
2. Wisconsin Electric Company letter NPL 99-0369, "TECHNICAL SPECIFICATION CHANGE REQUEST 210 AMENDMENT TO FACILITY OPERATING LICENSES TO REFLECT REQUIRED CHANGES TO THE TECHNICAL SPECIFICATIONS AS A RESULT OF USING UPGRADED FUEL POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 " from Mark E. Reddemann to Document Control Desk, dated June 22, 1999

3. Wisconsin Electric Company letter NPL 99-0731, "Supplement 1 to Technical Specification Change Request 210, Point Beach Nuclear Plant, Units 1 and 2," from Mark E. Reddemann to Document Control Desk, dated December 17, 1999
4. NRC Letter, Ashok Thadani to S.R. Tritch, "Acceptance for Referencing Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report',", July, 1991
5. Westinghouse "Reload Transition Safety Report for the Kewaunee Nuclear Power Plant," dated July 2002
6. WCAP–9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," by S. L. Davidson, et al, dated July 1985
7. WPSC Letter NRC 91-084 from K.H. Evers to U.S. Nuclear Regulator Commission Document Control Desk, "Core Reloads of Advanced Design Fuel Assemblies", dated June 19, 1991
8. Kewaunee Nuclear Power Plant, Reload Safety Evaluation Cycle 25, November 2001, Nuclear Management Company, LLC, Wisconsin Public Power Corporation, Wisconsin Power & Light
9. LTR-NRC-02-18, Submittal of WCAP-12488-A, Addendum 2/WCAP-14204-A, Addendum 2 of Westinghouse Fuel Criteria Evaluation Process, Revision to Design Criteria", April 26, 2002

ATTACHMENT 2

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 187

Strike-Out Pages for Technical Specifications including Bases

TS ii
TS vi
TS 1.0-6
TS 2.1-1
TS 2.3-2 through TS 2.3-3
TS 3.1-6
TS 3.4-3
TS 3.10-1 through TS 3.10-5
TS 3.10-9
TS Figure 3.1-3
TS 5.2-1
TS 5.3-1
TS 6.9-4
TS B2.1-1
TS B2.2-1
TS B3.1-8
TS B3.4-1 through TS B3.4-4
TS B3.10-2 through TS B3.10-5
TS B3.10-8

Section	Title	Page

LIST OF FIGURES

Note:

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

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated based on dose conversion factors derived from ICRP-30.~~with the methodology established in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."~~

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.~~0361~~0059	I-132
0.~~2703~~1692	I-133
0.~~0169~~0010	I-134
0.~~0838~~0293	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

b. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the program temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation

c. The peak fuel centerline temperature shall be maintained < 4700 °F

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, % RATED POWER

T = ~~Average temperature~~Reference Average Temperature at RATED POWER, °F

T' ≤= [*]°F

P = Pressurizer pressure, psig

P' = [*] psig

K_1 = [*]

K_2 = [*]

K_3 = [*]

τ_1 = [*] sec.

τ_2 = [*] sec.

$f(\Delta I)$ = An even function of the indicated difference between top and bottom detectors of the powerrange nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, % RATED POWER

T = ~~Average Temperature~~Reference Average Temperature at RATED POWER, °F

T' ≤=[*]°F

K_4 ≤ [*]

K_5 ≥ [*] for increasing T; [*] for decreasing T

K_6 ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f(\Delta I)$ = ~~As defined above~~0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level ≥ 5% of narrow range instrument span.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

   A. ≤ ~~20~~ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

   A. With the specific activity of the reactor coolant > ~~0.20~~1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding ~~the limit shown on Figure TS 3.1-3~~60 µCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

   B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

   C. With the specific activity of the reactor coolant > ~~0.20~~1.0 µCi/gram DOSE EQUIVALENT I-131 $or > \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of 39,000 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum of 39,000 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is ≤ 0.1 μCi/~~cc~~gram.

2. When the Reactor Coolant System temperature is > 350°F, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/~~cc~~ gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure: 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the SHUTDOWN MARGIN shall be at least that as specified in the COLR

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ exceeds its limit:

   A. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within the next 72 hours.

   B. Within 24 hours of initially being outside the limit, verify through flux mapping that $F_{\Delta H}^N$ has been restored to within the limit, or reduce thermal power to < 5% of RATED POWER within the next 2 hours, and

   C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power operation may proceed provided that $F_{\Delta H}^N$ is demonstrated, through incore flux mapping, to be within its limit prior to exceeding the following thermal power levels:

      i. A nominal 50% of RATED POWER,
      ii. A nominal 75% of RATED POWER, and
      iii. Within 24 hours of attaining ≥ 95% of RATED POWER

~~3~~3. If $F_Q^N(Z)$ exceeds its limit:

A. Reduce the thermal power at least 1% for each 1% $F_Q^N(Z)$ exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower Δ T Trip Setpoints within the next 72 hours by at least 1% for each 1% $F_Q^N(Z)$ exceeds its limit.

B. Identify and correct the cause of the out-of–limit condition prior to increasing thermal power above the reduced thermal power limit required by action A, above. Thermal power may be increased provided $F_Q^N(Z)$ is demonstrated, through incore flux mapping to be within its limit.

~~2.If, for any measured hot channel factor, the relationships of $F_Q^N(Z)$ and $F_{\Delta H}^N$ specified in the COLR are not true, then reactor power shall be reduced by a fractional amount of the design power to a value for which the relationships are true, and the high neutron flux trip setpoint shall be reduced by the same fractional amount. If subsequent incore mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, then the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.~~

~~3.~~4. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied.

5. ~~4.~~The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. ~~5.~~Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

A. During the comparison of incore to axial flux difference or ~~target flux difference determination or~~ once per effective full-power monthly interval, whichever occurs first.

B. Upon achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.5.A.

C. If a power distribution map indicates an increase in peak pin power, F_Q^N, of 2% or more, due to exposure, when compared to the last power distribution map, then either of the following actions shall be taken:

i. $F_Q^{EQ}(Z)$ shall be increased by the penalty factor specified in the COLR for comparison to the relationship of $F_Q^{EQ}(Z)$ specified in the COLR, or

ii. $F_Q^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every seven effective full-power days until a power distribution map indicates that the peak pin power, F_Q^N, is not increasing with exposure when compared to the last power distribution map.

~~6.~~7. If, for a measured F_Q^{EQ}, the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are not satisfied~~ and the relationships of $F_Q^N(Z)$ and $F_{\Delta H}^N$ specified in the COLR are satisfied~~, then ~~within 12 hours~~ take ~~one of~~ the following actions:

A. Reduce the axial flux difference limit at least 1% for each 1% $F_Q^{EQ}(Z)$ exceeds its limit within 4 hours after each determination and reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower Δ T Trip Setpoints within the 72 hours by at least 1% for each 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. Confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied prior to increasing thermal power above the maximim allowable power of the axial flux difference limits.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, within 15 minutes restore to within limits. If this action and associated completion time is not met, reduce thermal power to less than 50% RATED POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints ≤ 55 percent of RATED POWER within the next 72 hours.

B. If the alarms used to monitor the axial flux difference are rendered inoperable, verify that the axial flux difference is within limits for each operable excore channel once within one hour and every hour thereafter.

~~A. Take corrective actions to improve the power distribution and upon achieving equilibrium conditions measure the target flux difference and verify that the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are satisfied,~~

~~OR~~

~~B. Reduce reactor power and the high neutron flux trip setpoint by 1% for each percent that the left hand sides of the relationships of $F_Q^{EQ}(Z)$ specified in the COLR exceed the limits specified in the right hand sides. Reactor power may subsequently be increased provided that a power distribution map verifies that the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are satisfied with at least 1% of margin for each percent of power level to be increased.~~

~~7. The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per full power month.~~

~~8. The indicated axial flux difference shall be considered outside of the limits of TS 3.10.b.9 through TS 3.10.b.12 when more than one of the OPERABLE excore channels are indicating the axial flux difference to be outside a limit.~~

~~9. Except during physics tests, during excore detector calibration and except as modified by TS 3.10.b.10 through TS 3.10.b.12, the indicated axial flux difference shall be maintained within the target band about the target flux difference as specified in the COLR.~~

~~10. At a power level > 90% of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band within 15 minutes or reactor power shall be reduced to a level no greater than 90% of rated power.~~

~~11. At power levels > 50% and ≤ 90% of rated power:~~

~~A. The indicated axial flux difference may deviate from the target band, specified in the COLR, for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the outer envelope specified in the COLR. If the cumulative time exceeds one hour, then the reactor power shall be reduced to ≤50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤ 55% of rated power.~~

~~If the indicated axial flux difference exceeds the outer envelope specified in the COLR, then the reactor power shall be reduced to ≤50% of rated thermal power within 30 minutes and the high neutron flux setpoint reduced to ≤55% of rated power.~~

~~B. A power increase to a level > 90% of rated power is contingent upon the indicated axial flux difference being within its target band.~~

~~12. At a power level no greater than 50% of rated power:~~

~~A. The indicated axial flux difference may deviate from its target band.~~

~~B. A power increase to a level > 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) of the preceding 24-hour period.~~

~~One half of the time the indicated axial flux difference is out of its target band, up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level ≤90% of rated power.~~

~~13. Alarms shall normally be used to indicate nonconformance with the flux difference requirement of TS 3.10.b.10 or the flux difference time requirement of TS 3.10.b.11.A. If the alarms are temporarily out of service, then the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.~~

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within two hours:

   A. Eliminate the tilt.

   B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, then reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ ~~93,000~~178,000 gallons per minute average ~~per loop~~ and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, ~~between 70%~~at or above 90% ~~and 95%~~ power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

TS FIGURE TS 3.1-3
DELETED

5.2 CONTAINMENT

APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

OBJECTIVE

To define the significant design features of the Containment System.

SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the design basis accident are within the guidelines of 10 CFR Part ~~100~~50.67. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel reactor containment vessel designed for the peak pressure of the design basis accident.

   B. A concrete shield building which surrounds the containment vessel, providing a shield building annulus between the two structures.

   C. A Shield Building Ventilation System that causes leakage from the reactor containment vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System that serves the special ventilation zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

5.3 REACTOR CORE

APPLICABILITY

Applies to the reactor core.

OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy, ZIRLO™, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) 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, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(11) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(13) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES-TOPICAL REPORT," September 1974 (Westinghouse Proprietary).

(14) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure 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. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

~~Two~~ Three departure from nucleate boiling ratio (DNBR) correlations used in the safety analyses: the WRB-1 DNBR correlation, the high thermal performance (HTP) DNBR correlation and the W-3 DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The W-3 correlation is used when the coolant conditions are outside the range of the WRB-1 correlation or for the analysis of non-HTP FRA-ANP fuel designs and for all fuel designs at low pressure and temperature conditions (e.g., the conditions analyzed during a main steam line break accident). ~~Both~~ DNBR correlations have been qualified and approved for application to Kewaunee. The DNBR~~minimum~~ limits are 1.14 for the HTP correlation, 1.17 for the WRB-1correlation, and 1.30 for the W-3 correlation.

BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig~~.~~ for all transients except the hypothetical RCCA Ejection accident, for which the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia) is applied. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

---

[1] USAR Section 4
[2] USAR Section 4.3

Maximum Coolant Activity (TS 3.1.c)

The maximum dose ~~to the thyroid and whole body~~ that an individual may receive following an accident is specified in GDC 19 and 10 CFR ~~100~~50.67. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR ~~100~~50.67 limits.

The Reactor Coolant Specific Activity is limited to ≤ ~~0.2~~1.0 μCi/gram DOSE EQUIVALENT I-131 to ensure the ~~thyroid~~ dose does not exceed the GDC-19 and 10 CFR ~~100~~50.67 guidelines. The applicable accidents identified in the USAR[15] are analyzed assuming an RCS activity of ~~0.2~~1.0 μCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. -To ensure the conditions allowed ~~by Figure TS 3.1-3~~ are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of 60 μCi/gram DOSE EQUIVALENT I-131. ~~based on Figure TS 3.1-3.~~ The results obtained from these analyses indicate that the control room and off-site ~~thyroid~~ doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR ~~100~~50.67 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site ~~whole body~~ dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR ~~100~~50.67 limits.

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

---

[15] USAR Section 14.0

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure~~. The maximum full-power steam flow at 1721 1780 MWt is 7,449,0007,760,000 lbs./hr~~. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-heat-sink event. ~~Therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary.~~

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on an evaluation~~analyses~~ performed by ~~WPSC and~~ the Westinghouse Electric Company, LLC~~Corporation~~. This evaluation~~These analyses conservatively assumed the plant was at 100% initial power and~~ demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident below 15% of RATED POWER with no steam generator dryout, or ~~core damage~~reactor coolant system overpressure. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER~~ to provide further margin in the analysis~~.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses to fulfill the above functions. Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. ~~However, analyses from full power demonstrate that initiation of flow can be delayed for at least 10 minutes with no steam generator dryout or core damage.~~ The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

~~Analyses by WPSC and the Westinghouse Electric Corporation show that AFW-2A and AFW-2B may be in the throttled or closed position, or the AFW pump control switches located in the control room may be in the "pull out" position without a compromise to safety. This does not constitute a condition of inoperability as listed in TS 3.4.b.1 or TS 3.4.b.2. The analysis shows that diverse automatic reactor trips ensure a plant trip before any core damage or system overpressure occurs and that at least 10 minutes are available for the operators to manually initiate auxiliary feedwater flow (start AFW pumps or fully open AFW-2A and AFW-2B) for any credible accident from an initial power of 100%.~~
The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. ~~Although a~~nalyses have demonstrated that operation with the cross-connect valves closed is acceptable when reacctor power is~~, the TS restrict operation with the valves closed to~~ <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

---

[1] USAR Section 8.2.4

Secondary Activity Limits (TS 3.4.d)

The maximum dose ~~to the thyroid and whole body~~ that an individual may receive following an accident is specified in GDC 19 and 10 CFR ~~100~~ 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR ~~100~~50.67 limits.

The secondary side of the steam generator's activity is limited to ≤ 0.1 μCi/~~cc~~gram DOSE EQUIVALENT I-131 to ensure the ~~thyroid~~ dose does not exceed the GDC-19 and 10 CFR ~~100~~ 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/~~cc~~ gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site ~~thyroid dose~~ doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR ~~100~~ 50.67 limits.

[2] USAR Section 14.0

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured $F_Q^N(Z)$ obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures, with a high probability, remain less than the 2200° F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. The LOCA analyses are performed for Westinghouse 422 V-+ fuel, FRA-ANP heavy fuel and for FRA-ANP standard fuel.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. The safety analyses are performed for FRA-ANP heavy fuel, ~~and for~~ FRA-ANP standard fuel, and Westinghouse 422 V-+ fuel. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is less than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^N$ limit.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects need be included in TS 3.10.b.1 for FRA-ANP fuel.[1]

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. ~~4.~~ The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for~~m~~ monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and the reactor power is greater than 50 percent of RATED POWER.~~Axial power distribution control specifications which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.~~

[1]N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

~~The specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load follow maneuvers.[1]~~

~~Conformance with TS 3.10.b.9 through TS 3.10.b.12 ensures the F_Q^N upper bound envelope is not exceeded and xenon distributions will not develop which at a later time would cause greater local power peaking.~~

~~At the beginning of cycle, power escalation may proceed without the constraints of TS 3.10.b.5 since the startup test program provides adequate surveillance to ensure peaking factor limits. Target flux difference surveillance is initiated after achieving equilibrium conditions for sustained operation.~~

~~The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is determined from the nuclear instrumentation. This value, divided by the fraction of full power at which the core was OPERATING is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviations as specified in the COLR are permitted from the indicated reference value.~~

~~Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.~~

~~In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of 1 hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band.~~

~~The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference envelope as specified in the COLR. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the target band as specified in the COLR for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated.~~

~~As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.~~

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic Protection System. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, then the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

Core Average Temperature (TS 3.10.k)

The RCS core average temperature limit is consistent with full power operation within the nominal operational envelope. Either Tavg control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A higher Tavg will cause the reactor core to approach DNB limits.

Reactor Coolant System Pressure (TS 3.10.m)

The RCS pressure limit is consistent with operation within the nominal operational envelope. Either pressurizer pressure control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A lower pressure will cause the reactor core to approach DNB limits.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant system (RCS) flow limit, as specified in the COLR, is consistent with the minimum RCS flow limit assumed in the safety analysis adjusted by the measurement uncertainty. The safety analysis assumes initial conditions for plant parameters within the normal steady state envelope. The limits placed on the RCS pressure, temperature, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed tansients.

The RCS flow normally remains constant during an operational fuel cycle with all reactor coolant pumps running. At least two plant computer readouts from the loop RCS flow instrument channels are averaged per reactor coolant loop and the sum of the reactor coolant loop flows are compared to the limit. Operating within this limit will result in meeting the DNBR criterion in the event of a DNB-limited event.

ATTACHMENT 3

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 187

Revised Pages for Technical Specifications including Bases

TS ii
TS vi
TS 1.0-6
TS 2.1-1
TS 2.3-2 through TS 2.3-3
TS 3.1-6
TS 3.4-3
TS 3.10-1 through TS 3.10-3
TS 3.10-6
TS Figure 3.1-3
TS 5.2-1
TS 5.3-1
TS 6.9-4
TS B2.1-1
TS B2.2-1
TS B3.1-1
TS B3.1-8
TS B3.4-1 through TS B3.4-4
TS B3.10-2
TS B3.10-3
TS B3.10-6 through TS B3.10-7

Section	Title	Page

LIST OF FIGURES

Note:

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

p. DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 is that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be as listed and calculated based on dose conversion factors derived from ICRP-30.

DOSE CONVERSION FACTOR	ISOTOPE
1.0000	I-131
0.0059	I-132
0.1692	I-133
0.0010	I-134
0.0293	I-135

q. CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle-specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.a.4. Plant operation within these limits is addressed in individual Specifications.

r. SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means (TS 3.10.e), it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

b. In the OPERATING and HOT STANDBY MODES, the fuel and moderator temperatures are changed to the program temperature.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

APPLICABILITY

Applies to the limiting combination of thermal power, Reactor Coolant System pressure and coolant temperature during the OPERATING and HOT STANDBY MODES.

OBJECTIVE

To maintain the integrity of the fuel cladding.

SPECIFICATION

a. The combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits specified in the COLR. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

b. The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.14 for the HTP DNB correlation and 1.17 for the WRB-1 DNB correlation

c. The peak fuel centerline temperature shall be maintained < 4700 °F

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \left[K_1 - K_2 (T - T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 (P - P') - f(\Delta I) \right]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, % RATED POWER
T	=	Reference Average Temperature at RATED POWER, °F
T'	≤	[*]°F
P	=	Pressurizer pressure, psig
P'	=	[*] psig
K_1	=	[*]
K_2	=	[*]
K_3	=	[*]
τ_1	=	[*] sec.
τ_2	=	[*] sec.
$f(\Delta I)$	=	An even function of the indicated difference between top and bottom detectors of the powerrange nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

1. For $q_t - q_b$ within [*], [*] %, $f(\Delta I) = 0$.

2. For each percent that the magnitude of $q_t - q_b$ exceeds [*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

3. For each percent that the magnitude of $q_t - q_b$ exceed -[*] % the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] % of RATED POWER.

Note: [*] As specified in the COLR

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f (\Delta I) \right]$$

where

ΔT_0 = Indicated ΔT at RATED POWER, % RATED POWER

T = Reference Average Temperature at RATED POWER, °F

T' ≤ [*]°F

K_4 ≤ [*]

K_5 ≥ [*] for increasing T; [*] for decreasing T

K_6 ≥ [*] for T > T'; [*] for T < T'

τ_3 = [*] sec.

$f(\Delta I)$ = 0 for all ΔI

Note: [*] As specified in the COLR

4. Reactor Coolant Flow

A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

B. Reactor coolant pump motor breaker open

1. Low frequency setpoint ≥ 55.0 Hz

2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

Low-low steam generator water level ≥ 5% of narrow range instrument span.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

   A. ≤ 1.0 µCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

   A. With the specific activity of the reactor coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding 60 µCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

   B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

   C. With the specific activity of the reactor coolant > 1.0 µCi/gram DOSE EQUIVALENT I-131 $or > \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

c. Condensate Storage Tank

1. The Reactor Coolant System shall not be heated > 350°F unless a minimum of 39,000 gallons of water is available in the condensate storage tanks.

2. If the Reactor Coolant System temperature is > 350°F and a minimum of 39,000 gallons of water is not available in the condensate storage tanks, reactor operation may continue for up to 48 hours.

3. If the time limit of TS 3.4.c.2 above cannot be met, within 1 hour initiate action to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

d. Secondary Activity Limits

1. The Reactor Coolant System shall not be heated > 350°F unless the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators is ≤ 0.1 μCi/gram.

2. When the Reactor Coolant System temperature is > 350°F, the DOSE EQUIVALENT Iodine-131 activity on the secondary side of the steam generators may exceed 0.1 μCi/gram for up to 48 hours.

3. If the requirement of TS 3.4.d.2 cannot be met, then within 1 hour action shall be initiated to:

   - Achieve HOT STANDBY within 6 hours
   - Achieve HOT SHUTDOWN within the following 6 hours
   - Achieve and maintain the Reactor Coolant System temperature < 350°F within an additional 12 hours.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

APPLICABILITY

Applies to the limits on core fission power distributions and to the limits on control rod operations.

OBJECTIVE

To ensure: 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

SPECIFICATION

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the SHUTDOWN MARGIN shall be at least that as specified in the COLR

b. Power Distribution Limits

1. At all times, except during Low Power Physics Tests, the hot channel factors defined in the basis must meet the following limits:

   A. $F_Q^N(Z)$ Limits shall be as specified in the COLR.
   B. $F_{\Delta H}^N$ Limits shall be as specified in the COLR.

2. If $F_{\Delta H}^N$ exceeds its limit:

   A. Within 4 hours either, restore $F_{\Delta H}^N$ to within its limit or reduce thermal power to less than 50% of RATED POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED POWER within the next 72 hours.

   B. Within 24 hours of initially being outside the limit, verify through flux mapping that $F_{\Delta H}^N$ has been restored to within the limit, or reduce thermal power to < 5% of RATED POWER within the next 2 hours, and

   C. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A and/or B, above. Subsequent power operation may proceed provided that $F_{\Delta H}^N$ is demonstrated, through incore flux mapping, to be within its limit prior to exceeding the following thermal power levels:

      i. A nominal 50% of RATED POWER,
      ii. A nominal 75% of RATED POWER, and
      iii. Within 24 hours of attaining ≥ 95% of RATED POWER

3. If $F_Q^N(Z)$ exceeds its limit:

   A. Reduce the thermal power at least 1% for each 1% $F_Q^N(Z)$ exceeds its limit within 15 minutes after each determination and similarly reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower Δ T Trip Setpoints within the next 72 hours by at least 1% for each 1% $F_Q^N(Z)$ exceeds its limit.

   B. Identify and correct the cause of the out-of-limit condition prior to increasing thermal power above the reduced thermal power limit required by action A, above. Thermal power may be increased provided $F_Q^N(Z)$ is demonstrated, through incore flux mapping to be within its limit.

4. Following initial loading and at regular effective full-power monthly intervals thereafter, power distribution maps using the movable detection system shall be made to confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied.

5. The measured $F_Q^{EQ}(Z)$ hot channel factors under equilibrium conditions shall satisfy the relationship for the central axial 80% of the core as specified in the COLR.

6. Power distribution maps using the movable detector system shall be made to confirm the relationship of $F_Q^{EQ}(Z)$ specified in the COLR according to the following schedules with allowances for a 25% grace period:

   A. During the comparison of incore to axial flux difference or once per effective full-power monthly interval, whichever occurs first.

   B. Upon achieving equilibrium conditions after reaching a thermal power level > 10% higher than the power level at which the last power distribution measurement was performed in accordance with TS 3.10.b.5.A.

   C. If a power distribution map indicates an increase in peak pin power, F_Q^N, of 2% or more, due to exposure, when compared to the last power distribution map, then either of the following actions shall be taken:

      i. $F_Q^{EQ}(Z)$ shall be increased by the penalty factor specified in the COLR for comparison to the relationship of $F_Q^{EQ}(Z)$ specified in the COLR, or

      ii. $F_Q^{EQ}(Z)$ shall be measured by power distribution maps using the incore movable detector system at least once every seven effective full-power days until a power distribution map indicates that the peak pin power, F_Q^N, is not increasing with exposure when compared to the last power distribution map.

7. If, for a measured F_Q^{EQ}, the relationships of $F_Q^{EQ}(Z)$ specified in the COLR are not satisfied, then take the following actions:

   A. Reduce the axial flux difference limit at least 1% for each 1% F_Q^{EQ} (Z) exceeds its limit within 4 hours after each determination and reduce the Power Range Neutron Flux-High Trip Setpoints and the Overpower Δ T Trip Setpoints within the 72 hours by at least 1% for each 1% that the maximum allowable power of the axial flux difference limits is reduced.

B. Confirm that the hot channel factor limits of TS 3.10.b.1 are satisfied prior to increasing thermal power above the maximim allowable power of the axial flux difference limits.

8. Axial Flux Difference

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

A. During power operation with thermal power ≥ 50 percent of RATED POWER, the axial flux difference shall be maintained within the limits specified in the COLR.

i. If the axial flux difference is not within limits, within 15 minutes restore to within limits. If this action and associated completion time is not met, reduce thermal power to less than 50% RATED POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints ≤ 55 percent of RATED POWER within the next 72 hours.

B. If the alarms used to monitor the axial flux difference are rendered inoperable, verify that the axial flux difference is within limits for each operable excore channel once within one hour and every hour thereafter.

c. Quadrant Power Tilt Limits

1. Except for physics tests, whenever the indicated quadrant power tilt ratio > 1.02, one of the following actions shall be taken within two hours:

A. Eliminate the tilt.

B. Restrict maximum core power level 2% for every 1% of indicated power tilt ratio > 1.0.

2. If the tilt condition is not eliminated after 24 hours, then reduce power to 50% or lower.

3. Except for Low Power Physics Tests, if the indicated quadrant tilt is > 1.09 and there is simultaneous indication of a misaligned rod:

A. Restrict maximum core power level by 2% of rated values for every 1% of indicated power tilt ratio > 1.0.

B. If the tilt condition is not eliminated within 12 hours, then the reactor shall be brought to a minimum load condition (≤30 Mwe).

4. If the indicated quadrant tilt is > 1.09 and there is no simultaneous indication of rod misalignment, then the reactor shall immediately be brought to a no load condition (≤5% reactor power).

j. Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the quadrant tilt shall be logged once per shift and after a load change > 10% of rated power or after > 24 steps of control rod motion. The monitors shall be set to alarm at 2% tilt ratio.

k. Core Average Temperature

During steady-state power operation, T_{ave} shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

l. Reactor Coolant System Pressure

During steady-state power operation, Reactor Coolant System pressure shall be maintained within the limits specified in the COLR, except as provided by TS 3.10.n.

m. Reactor Coolant Flow

1. During steady-state power operation, reactor coolant total flow rate shall be ≥ 178,000 gallons per minute average and greater than or equal to the limit specified in the COLR. If reactor coolant flow rate is not within the limits as specified in the COLR, action shall be taken in accordance with TS 3.10.n.

2. Compliance with this flow requirement shall be demonstrated by verifying the reactor coolant flow during initial power escalation following each REFUELING, at or above 90% power with plant parameters as constant as practical.

n. DNBR Parameters

If, during power operation any of the conditions of TS 3.10.k, TS 3.10.l, or TS 3.10.m.1 are not met, restore the parameter in two hours or less to within limits or reduce power to < 5% of thermal rated power within an additional six hours. Following analysis, thermal power may be raised not to exceed a power level analyzed to maintain a DNBR greater than the minimum DNBR limit.

TS FIGURE TS 3.1-3
DELETED

5.2 CONTAINMENT

APPLICABILITY

Applies to those design features of the Containment System relating to operational and public safety.

OBJECTIVE

To define the significant design features of the Containment System.

SPECIFICATION

a. Containment System

1. The Containment System completely encloses the entire reactor and the Reactor Coolant System and ensures that leakage of activity is limited, filtered and delayed such that off-site doses resulting from the design basis accident are within the guidelines of 10 CFR Part 50.67. The Containment System provides biological shielding for both normal OPERATING conditions and accident situations.

2. The Containment System consists of:

   A. A free-standing steel reactor containment vessel designed for the peak pressure of the design basis accident.

   B. A concrete shield building which surrounds the containment vessel, providing a shield building annulus between the two structures.

   C. A Shield Building Ventilation System that causes leakage from the reactor containment vessel to be delayed and filtered before its release to the environment.

   D. An Auxiliary Building Special Ventilation System that serves the special ventilation zone and supplements the Shield Building Ventilation System during an accident condition by causing any leakage from the Residual Heat Removal System (RHRS) and certain small amounts of leakage that might be postulated to bypass the Shield Building Ventilation System to be filtered before their release.

5.3 REACTOR CORE

APPLICABILITY

Applies to the reactor core.

OBJECTIVE

To define those design features which are essential in providing for safe reactor core operations.

SPECIFICATION

a. Fuel Assemblies

The reactor shall contain 121 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy, ZIRLO™, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-test-assemblies that have not completed representative testing may be placed in non-limiting core regions. Lead-test-assemblies shall be of designs approved by the NRC for use in pressurized water reactors and their clad materials shall be the materials approved as part of those designs.

b. Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

(3) Nissley, M.E. et, al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, March 1991, Volume 1: Model Description and Validation; Addendum 4: Model Revisions.

(4) N. Lee et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-NP-A (Non-Proprietary), dated August 1985.

(5) C.M. Thompson, et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condesation Model," WCAP-10054-P-A, Addendum 2, Revision 1 (Proprietary) and WCAP-10081-NP (Non-Proprietary), dated July 1997.

(6) 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, dated October 1986.

(7) ANF-88-133 (P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Meghodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, dated December 1991.

(8) EMF-92-116 (P)(A) Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, dated February 1999.

(9) XN-NF-77-57, Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II, dated January 1978, and Supplement 2, dated October 1981.

(10) WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," February 1994 (W Proprietary).

(11) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary)

(12) WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

(13) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES-TOPICAL REPORT," September 1974 (Westinghouse Proprietary).

(14) WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

BASIS - Safety Limits-Reactor Core (TS 2.1)

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all OPERATING conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters of RATED POWER, reactor coolant temperature and pressure have been related to DNB through a DNB correlation. The DNB correlation has been developed to predict the DNB heat flux and the location of the DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to the DNBR limit. This minimum DNBR corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all OPERATING conditions.

The SAFETY LIMIT curves as provided in the Core Operating Report Limits Report which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure 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. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNBR ratio reaches the DNBR limit and thus, this limit is conservative with respect to maintaining clad integrity. The area where clad integrity is ensured is below these lines.

The curves are based on the nuclear hot channel factor limits of as specified in the COLR.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are given in TS 3.10.d. Slightly higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits as specified in the COLR ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and PROTECTION SYSTEM is designed to prevent any anticipated combination of transient conditions that would result in a DNBR less than the DNBR limit.

Three departure from nucleate boiling ratio (DNBR) correlations used in the safety analyses: the WRB-1 DNBR correlation, the high thermal performance (HTP) DNBR correlation and the W-3 DNBR correlation. The WRB-1 correlation applies to the Westinghouse 422 V+ fuel. The HTP correlation applies to FRA-ANP fuel with HTP spacers. The W-3 correlation is used when the coolant conditions are outside the range of the WRB-1 correlation or for the analysis of non-HTP FRA-ANP fuel designs and for all fuel designs at low pressure and temperature conditions (e.g., the conditions analyzed during a main steam line break accident). DNBR correlations have been qualified and approved for application to Kewaunee. The DNBR limits are 1.14 for the HTP correlation, 1.17 for the WRB-1correlation, and 1.30 for the W-3 correlation.

BASIS - Safety Limit - Reactor Coolant System Pressure (TS 2.2)

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is ensured. The maximum transient pressure allowable in the reactor pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USASI B.31.1.0 is 120% of design pressure. Thus, the SAFETY LIMIT of 2735 psig (110% of design pressure, 2485 psig) has been established.[2]

The settings of the power-operated relief valves, the reactor high pressure trip and the safety valves have been established to prevent exceeding the SAFETY LIMIT of 2735 psig for all transients except the hypothetical RCCA Ejection accident, for which the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia) is applied. The initial hydrostatic test was conducted at 3107 psig to ensure the integrity of the Reactor Coolant System.

---

[1] USAR Section 4
[2] USAR Section 4.3

BASIS - Reactor Coolant System (TS 3.1.a)

Reactor Coolant Pumps (TS 3.1.a.1)

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part one of the specification requires that both reactor coolant pumps be OPERATING when the reactor is in power operation to provide core cooling. Planned power operation with one loop out-of-service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this MODE of operation. The flow provided in each case in part one will keep Departure from Nucleate Boiling Ratio (DNBR) well above 1.34. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted except for tests. Upon loss of one pump below 10% full power, the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation can remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost.[1]

The RCS will be protected against exceeding the design basis of the Low Temperature Overpressure Protection (LTOP) System by restricting the starting of a Reactor Coolant Pump (RXCP) to when the secondary water temperature of each SG is < 100°F above each RCS cold leg temperature. The restriction on starting a reactor coolant pump (RXCP) when one or more RCS cold leg temperatures is ≤ 200°F is provided to prevent a RCS pressure transient, caused by an energy addition from the secondary system, which could exceed the design basis of the LTOP System.

Decay Heat Removal Capabilities (TS 3.1.a.2)

When the average reactor coolant temperature is ≤ 350°F a combination of the available heat sinks is sufficient to remove the decay heat and provide the necessary redundancy to meet the single failure criterion.

When the average reactor coolant temperature is ≤ 200°F, the plant is in a COLD SHUTDOWN condition and there is a negligible amount of sensible heat energy stored in the Reactor Coolant System. Should one residual heat removal train become inoperable under these conditions, the remaining train is capable of removing all of the decay heat being generated.

---

[1] USAR Section 7.2.2

Maximum Coolant Activity (TS 3.1.c)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is limited to ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[15] are analyzed assuming an RCS activity of 1.0 μCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of 60 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 50.67 limits.

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

---

[15] USAR Section 14.0

BASIS - Steam and Power Conversion System (TS 3.4)

Main Steam Safety Valves (TS 3.4.a)

The ten main steam safety valves (MSSVs) (five per steam generator) have a total combined rated capability of 7,660,380 lbs./hr. at 1181 lbs./in.^{2} pressure. This flow ensures that the main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by ASME B&PV Code) for the worst-case loss-of-heat-sink event.

While the plant is in the HOT SHUTDOWN condition, at least two main steam safety valves per steam generator are required to be available to provide sufficient relief capacity to protect the system.

The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Plan.

Auxiliary Feedwater System (TS 3.4.b)

The Auxiliary Feedwater (AFW) System is designed to remove decay heat during plant startups, plant shutdowns, and under accident conditions. During plant startups and shutdowns the system is used in the transition between Residual Heat Removal (RHR) System decay heat removal and Main Feedwater System operation.

The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow from the AFW pumps to the steam generators are OPERABLE. This requires that the two motor-driven AFW pumps be OPERABLE, each capable of taking suction from the Service Water System and supplying AFW to separate steam generators. The turbine-driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the main steam isolation valves and shall be capable of taking suction from the Service Water System and supplying AFW to both of the steam generators. With no AFW trains OPERABLE, immediate action shall be taken to restore a train.

Auxiliary feedwater trains are defined as follows:

"A" train -	"A" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "A" steam generator, not including AFW-10A or AFW-10B
"B" train -	"B" motor-driven auxiliary feedwater pump and associated AFW valves and piping to "B" steam generator, not including AFW-10A or AFW-10B
Turbine-driven train -	Turbine-driven AFW pump and associated AFW valves and piping to both "A" steam generator and "B" steam generator, including AFW-10A and AFW-10B

In the unlikely event of a loss of off-site electrical power to the plant, continued capability of decay heat removal would be ensured by the availability of either the steam-driven AFW pump or one of the two motor-driven AFW pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump and turbine-driven AFW pump is normally aligned to both steam generators. Valves AFW-10A and AFW-10B are normally open. Any single AFW pump can supply sufficient feedwater for removal of decay heat from the reactor.

As the plant is cooled down, heated up, or operated in a low power condition, AFW flow will have to be adjusted to maintain an adequate water inventory in the steam generators. This can be accomplished by any one of the following:

1. Throttling the discharge valves on the motor-driven AFW pumps

2. Closing one or both of the cross-connect flow valves

3. Stopping the pumps

If the main feedwater pumps are not in operation at the time, valves AFW-2A and AFW-2B must be throttled or the control switches for the AFW pumps located in the control room will have to be placed in the "pull out" position to prevent their continued operation and overfill of the steam generators. The cross-connect flow valves may be closed to specifically direct AFW flow. Manual action to re-initiate flow after it has been isolated is considered acceptable based on an evaluation performed by the Westinghouse Electric Company, LLC. This evaluation demonstrated that operators have at least 10 minutes to manually initiate AFW during any design basis accident below 15% of RATED POWER with no steam generator dryout, or reactor coolant system overpressure. The placing of the AFW control switches in the "pull out" position, the closing of one or both cross-connect valves, and the closing or throttling of valves AFW-2A and AFW-2B are limited to situations when reactor power is <15% of RATED POWER.

During accident conditions, the AFW System provides three functions:

1. Prevents thermal cycling of the steam generator tubesheet upon loss of the main feedwater pump

2. Removes residual heat from the Reactor Coolant System until the temperature drops below 300-350°F and the RHR System is capable of providing the necessary heat sink

3. Maintains a head of water in the steam generator following a loss-of-coolant accident

Each AFW pump provides 100% of the required capacity to the steam generators as assumed in the accident analyses to fulfill the above functions. Since the AFW System is a safety features system, the backup pump is provided. This redundant motor-driven capability is also supplemented by the turbine-driven pump.

The pumps are capable of automatic starting and can deliver full AFW flow within one minute after the signal for pump actuation. The head generated by the AFW pumps is sufficient to ensure that feedwater can be pumped into the steam generators when the safety valves are discharging and the supply source is at its lowest head.

The OPERABILITY of the AFW System following a main steam line break (MSLB) was reviewed in our response to IE Bulletin 80-04. As a result of this review, requirements for the turbine-driven AFW pump were added to the Technical Specifications.

For all other design basis accidents, the two motor-driven AFW pumps supply sufficient redundancy to meet single failure criteria. In a secondary line break, it is assumed that the pump discharging to the intact steam generator fails and that the flow from the redundant motor-driven AFW pump is discharging out the break. Therefore, to meet single failure criteria, the turbine-driven AFW pump was added to Technical Specifications.

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Analyses have demonstrated that operation with the cross-connect valves closed is acceptable when reacctor power is <15% of RATED POWER. At > 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an OPERATING condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for four hours of decay heat removal. The four hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met. [1]

[1] USAR Section 8.2.4

Secondary Activity Limits (TS 3.4.d)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on secondary coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The secondary side of the steam generator's activity is limited to $\leq$ 0.1 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[2] are analyzed assuming various inputs including steam generator activity of 0.1 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

[2] USAR Section 14.0

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor

$F_Q^N(Z)$, Height Dependent Nuclear Flux Hot Channel Factor, is defined as the maximum local linear power density in the core at core elevation Z divided by the core average linear power density, assuming nominal fuel rod dimensions.

$F_Q^{EQ}(Z)$ is the measured $F_Q^N(Z)$ obtained at equilibrium conditions during the target flux determination.

An upper bound envelope for $F_Q^N(Z)$ as specified in the COLR has been determined from extensive analyses considering all OPERATING maneuvers consistent with the Technical Specifications on power distribution control as given in TS 3.10. The results of the loss-of-coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures, with a high probability, remain less than the 2200 °F limit.

The $F_Q^N(Z)$ limits as specified in the COLR are derived from the LOCA analyses. The LOCA analyses are performed for Westinghouse 422 V+ fuel, FRA-ANP heavy fuel and for FRA-ANP standard fuel.

When a $F_Q^N(Z)$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

$F_Q^N(Z)$ is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the maximum integral of linear power along a fuel rod to the core average integral fuel rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in DNBR calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

The $F_{\Delta H}^N$ limit is determined from safety analyses of the limiting DNBR transient events. The safety analyses are performed for FRA-ANP heavy fuel, FRA-ANP standard fuel, and Westinghouse 422 V+ fuel. In these analyses, the important operational parameters are selected to minimize DNBR. The results of the safety analyses must demonstrate that minimum DNBR is less than the DNBR limit for a fuel rod operating at the $F_{\Delta H}^N$ limit.

The use of $F_{\Delta H}^N$ in TS 3.10.b.5.C is to monitor "upburn" which is defined as an increase in $F_{\Delta H}^N$ with exposure. Since this is not to be confused with observed changes in peak power resulting from such phenomena as xenon redistribution, control rod movement, power level changes, or changes in the number of instrumented thimbles recorded, an allowance of 2% is used to account for such changes.

Rod Bow Effects

No penalty for rod bow effects need be included in TS 3.10.b.1 for FRA-ANP fuel.[1]

Surveillance

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identifies operational anomalies which would otherwise affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than an indicated 12 steps from the bank demand position where reactor power is ≥85%, or an indicated 24 steps when reactor power is < 85%.

2. Control rod banks are sequenced with overlapping banks as specified in the COLR.

3. The control bank insertion limits as specified in the COLR are not violated, except as allowed by TS 3.10.d.2.

4. The axial power distribution, expressed in terms of axial flux difference, is maintained within the limits.

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the FQ(Z) upper bound envelope of FQLIMIT times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor program. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and the reactor power is greater than 50 percent of RATED POWER.

For Condition II events the core is protected from overpower and a minimum DNBR less than the DNBR limit by an automatic Protection System. Compliance with the specification is assumed as a precondition for Condition II transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

---

[1]N. E. Hoppe, "Mechanical Design Report Supplement for Kewaunee High Burnup (49 GWD/MTU) Fuel Assemblies," XN-NF-84-28(P), Exxon Nuclear Company, July 1984.

Inoperable Rod Position Indicator Channels (TS 3.10.f)

The rod position indicator channel is sufficiently accurate to detect a rod ± 12 steps away from its demand position. If the rod position indicator channel is not OPERABLE, then the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry.

Inoperable Rod Limitations (TS 3.10.g)

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30-day period is provided for the reanalysis of all accidents sensitive to the changed initial condition.

Rod Drop Time (TS 3.10.h)

The required drop time to dashpot entry is consistent with safety analysis.

Core Average Temperature (TS 3.10.k)

The RCS core average temperature limit is consistent with full power operation within the nominal operational envelope. Either Tavg control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A higher Tavg will cause the reactor core to approach DNB limits.

Reactor Coolant System Pressure (TS 3.10.m)

The RCS pressure limit is consistent with operation within the nominal operational envelope. Either pressurizer pressure control board indicator readings or computer indications are averaged to obtain the value for comparison to the limit. The limit is based on the average of either 4 control board indicator readings or 4 computer indications. A lower pressure will cause the reactor core to approach DNB limits.

Reactor Coolant Flow (TS 3.10.m)

The reactor coolant system (RCS) flow limit, as specified in the COLR, is consistent with the minimum RCS flow limit assumed in the safety analysis adjusted by the measurement uncertainty. The safety analysis assumes initial conditions for plant parameters within the normal steady state envelope. The limits placed on the RCS pressure, temperature, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the analyzed tansients.

The RCS flow normally remains constant during an operational fuel cycle with all reactor coolant pumps running. At least two plant computer readouts from the loop RCS flow instrument channels are averaged per reactor coolant loop and the sum of the reactor coolant loop flows are compared to the limit. Operating within this limit will result in meeting the DNBR criterion in the event of a DNB-limited event.

DNBR Parameters (TS 3.10.n)

The DNBR related safety analyses make assumptions on reactor temperature, pressure, and flow. In the event one of these parameters does not meet the TS 3.10.k, TS 3.10.l or TS 3.10.m limits, an analysis can be performed to determine a power level at which the MDNBR limit is satisfied.

ATTACHMENT 4

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

July 26, 2002

License Amendment Request 187

Westinghouse Report
Technical Design Basis for the Transition to 422V+ Fuel

Attachment 4

Technical Design Basis for the Transition to 422V+Fuel

Technical Design Basis for the Transition to 422V+Fuel

Technical Design Basis for the Transition to 422V+Fuel

List of Tables

Technical Design Basis for the Transition to 422V+Fuel

List of Figures

1 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Kewaunee Nuclear Power Plant (KNPP) plans to refuel and operate with upgraded Westinghouse fuel features commencing with Cycle 26. The upgraded fuel is 0.422-inch outside diameter (OD), 14x14, VANTAGE + fuel with PERFORMANCE + features, hereafter referred to as 422V+ fuel. This fuel that is currently operating in KNPP Cycle 25 as lead use assemblies (LUAs) is similar to the 422V+ fuel assemblies in their second cycle of operation in Point Beach Units 1 and 2.

This report summarizes the safety evaluations and analyses that were performed to confirm the acceptability of 422V+ fuel for KNPP operations. Sections 2.0 through 6.0 of this report address mechanical design features, nuclear design, thermal-hydraulic design, accident analyses, (loss-of-coolant accident [LOCA], non-LOCA, containment, radiological), and systems and component analyses, respectively. The analyses and evaluations included in this report support the Technical Specification changes requested herein and the changes to the KNPP Updated Safety Analysis Report (USAR) (Reference 1). Chapter 14 "Accident Analysis" that support the fuel upgrade (FU) are included in this submittal. It should be noted that this submittal is only for approval of the FU. Although analyses addressed herein have been performed at uprated (PU) conditions, a separate license amendment request for power uprating (PU) will be submitted at a later date.

This license amendment request serves as a reference safety evaluation and analysis report for the region-by-region reload transition from the KNPP Cycle 25 core to subsequent cores containing 422V+ fuel. Thus, the analysis results establish the new record of analysis applicable to future KNPP reload cores, through transition cycles, with the upgraded 422V+ fuel features.

Key safety parameters for the analyses have been chosen to maximize the applicability of the analysis results to future reload cycle evaluations which will be performed using the Westinghouse standard reload methodology (Reference 2). The objective of subsequent cycle specific Reload Safety Evaluations (RSEs) will be to verify that the applicable safety limits are not exceeded based on the reference analyses currently in the USAR (Reference 1) or as established in this licensing amendment request.

1.2 UPGRADED FUEL FEATURES (422V+)

KNPP Cycle 26 and subsequent cores will have fuel assemblies that incorporate: the 0.422-inch OD fuel rods; an optimized fuel assembly (OFA) style low-pressure drop (LPD), ZIRLO™ mid-grid for the 0.422-inch OD rod; ZIRLO fuel cladding; ZIRLO fabricated guide thimbles and instrumentation tubing; and mid-enriched annular pellets in axial blankets. The 0.422-inch OD fuel rod and associated mid-grid are features being incorporated for KNPP. ZIRLO cladding, ZIRLO fabricated components and annular pellets in axial blankets are known as VANTAGE + features that have been submitted to the Nuclear Regulatory Commission (NRC) in the licensing topical report, "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610, Appendices A through D (Reference 3), Appendix E (Reference 4), Appendices F and G (Reference 5), and associated Addenda 1 through 4 (References 1 through 9) 422V+ has received generic NRC approval (Reference 10 and 11) for lead rod burnups up to 60,000 MWD/MTU. The mid-enrichment of the annular pellets in axial blankets and zirconium oxide

coated cladding are PERFORMANCE + features which were reviewed by Westinghouse in SECL-92-305 (Reference 12) under 10 CFR 50.59 guidelines, and do not require NRC approval.

The 422V+ fuel assembly skeleton is similar to that of the 14x14 OFA and 14x14 STANDARD fuel that have operated for several cycles in two-loop Westinghouse plants, except for modifications necessary to accommodate higher burnup levels (lead rod burnups beyond 60,000 MWD/MTU when licensed by the NRC) and those modifications necessary to accommodate the 0.422-inch OD fuel rod. Additional modifications consist of the use of ZIRLO clad fuel rods, ZIRLO guide thimbles and instrumentation tubes.

Since 422V+ fuel is intended to replace the Framatome/ANP fuel, the 422V+ exterior assembly envelope is similar in design dimensions (refer to Table 2-1 in Section 2), and the functional interface with the reactor internals is equivalent to those of the Framatome/ANP fuel for which KNPP is currently licensed. Also, the 422V+ fuel assembly is designed to be mechanically and hydraulically compatible with the Framatome/ANP design in transition cores, and the same functional requirements and design criteria previously established for the Westinghouse OFA (Reference 13) remain valid for the 422V+ fuel assembly. Table 2-1 compares the 422V+ fuel assembly to the current Framatome/ANP fuel design. In addition, the 422V+ fuel assembly is designed to be mechanically, thermal-hydraulically, and neutronically compatible with Framatome/ANP in spent fuel storage and refueling activities.

The 422V+ fuel rod design represents a modification to the current OFA fuel rod design or its predecessor, the STANDARD (STD) fuel rod design, with the use of ZIRLO fuel cladding in place of Zircaloy-4 cladding to support higher burnup levels. The 422V+ fuel rods will contain enriched uranium dioxide fuel pellets, Gadiolinia bearing fuel pellets, and mid-enriched annular pellets in axial blankets. The KNPP 422V+ fuel rod design, for lead rod burnups beyond 60,000 MWD/MTU, is based on the ZIRLO fuel performance models given in Reference 3 and as modified in Reference 14, which is currently pending generic NRC approval. Additional debris mitigation protection is provided on the 422V+ fuel assembly through the use of zirconium-oxide coated cladding which is a PERFORMANCE + feature.

1.3 PEAKING FACTORS

The full power $F^N_{\Delta H}$ peaking factor design limit is 1.70. The full power $F_Q^N(Z)$ peaking factor limit is 2.50. These values will permit flexibility in developing fuel management schemes for longer fuel cycles and improved fuel economy and neutron utilization.

1.4 RTDP UNCERTAINTIES

KNPP is currently licensed for the Standard Thermal Design Procedure (STDP) methodology. Evaluation of various plant parameter uncertainties is required as a result of the Fuel Upgrade/Power Uprating (FU/PU) Program. Power uprating, conversion to the Revised Thermal Design Procedure (RTDP), and the current regulatory environment have all been considered in assessing the need to evaluate various plant parameter uncertainties. The evaluation of uncertainties requires a review of temperature, pressure, power and flow uncertainties used in the safety analysis. The uncertainties are calculated based on installed plant instrumentation or special test equipment and on calibration and calorimetric procedures.

The method of uncertainty analysis is discussed in Reference 15 and is the same regardless of whether the application to the safety analysis is RTDP or non-RTDP methodology. The uncertainty analysis statistically combines the individual uncertainties using the square root of the sum of the squares (SRSS) method. The analysis includes uncertainties for 1) the method of measurement, 2) the type of field device (that is, RTDs, transmitters, special test measurements), and 3) the calibration of the instrumentation. The uncertainties for temperature, pressure, power and flow are then used in the development of the reactor core limits and the departure from nucleate boiling ratio (DNBR) limits. The ΔT reactor trip setpoints are then developed from the new core limits for use in the Technical Specifications. ΔT trip setpoints will be documented in the Core Operating Limits Report (COLR).

Not all analyses use RTDP methodology. For those analyses that do not use RTDP methodology, STDP methods are still employed. The difference between the two methodologies is in the initial conditions used in the analysis and the application of the uncertainties. For the RTDP events, the uncertainties are included in the development of the DNBR limit, and nominal values are assumed for the initial conditions for reactor coolant system (RCS) pressure, RCS temperature, and reactor power. Minimum measured flow (MMF), which is used with RTDP, is equivalent to the thermal design flow (TDF), which is used with STDP, plus a flow uncertainty. For those events using STDP methodology, the uncertainties are directly applied to the nominal values for RCS pressure, RCS temperature and power to define the initial conditions for the non-LOCA events. For events using RTDP methodology, the uncertainties are statistically combined with the DNBR correlation uncertainties to obtain the overall DNBR uncertainty factor used to define the design DNBR limit. Thus, nominal values for RCS pressure, RCS temperature and power are used for the initial conditions for the non-LOCA events. This methodology is consistent with the reference licensing basis analyses found in the KNPP USAR (Reference 1). Whether positive or negative, uncertainties are applied in a manner that is consistent with the analysis and is in the most conservative direction for a specific event. Analyses that use STDP and those that use RTDP are delineated in Chapter 14 of the KNPP USAR.

Table 1-1 is a summary of the RTDP uncertainties used by all analysis groups. Also listed are the calculated RTDP uncertainties. It can be seen that the uncertainties for the FU/PU Program have increased. It should be noted that the analysis uncertainties are actually slightly larger than those calculated during the RTDP uncertainty analysis. The rationale of using slightly larger values for the uncertainties ensures conservatism in the overall analysis.

1.5 PERFORMANCE CAPABILITIES WORKING GROUP PARAMETERS

The analyses for the KNPP FU/PU Program are based on parameters specified in the Performance Capabilities Working Group (PCWG) parameter sheet. This parameter sheet is used by all analysis groups to ensure consistent use of parameters for the analyses. The parameter sheet is provided in Table 1-2. This sheet includes parameters for four cases at uprated power conditions. Two cases involve parameters at the low end of the T_{avg} window (the full range of average core temperature used to support the safety analysis and component evaluations) for 0 percent and 10 percent steam generator tube plugging, and two cases involve parameters at the upper end of the T_{avg} window for 0 percent and 10 percent steam generator tube plugging. The 10 percent tube plugging limit represents the maximum allowable tube plugging for a single steam generator.

1.6 GENERAL ANALYSIS ASSUMPTIONS

One purpose of the re-analysis performed for the KNPP FU/PU Program is to update and confirm many of the assumptions and inputs used in the analyses. These new or revised assumptions and input parameters form the basis upon which the analyses are performed and ultimately establish the KNPP licensing basis (that is, Technical Specifications and USAR analyses of record). The process begins by Westinghouse documenting the assumptions and input parameters originally expected to be used in the analyses. Nuclear Management Company (NMC) reviews, updates and approves the list. When concurrence is obtained, the assumptions and input parameters are documented as final values which are used in the analyses.

The use of uprated power and trip setpoints based on the uprated power requires comment. The uprated nuclear steam supply system (NSSS) power of 1780 MWt (1772 MWt core power) is bounding for analyses documented herein. During the transition period, which begins when the new analyses become the analyses of record and ends when the uprate amendment request is approved, the 1780 MWt analysis basis will bound the current NSSS power level of 1657.1 MWt (1650 MWt core power). The reactor trip setpoints will also be bounding since they were generated for the uprated power level, but will be used for the lower operating NSSS power level of 1657.1 MWt. They are, therefore, more restrictive than the current trip setpoints that are being used for the current NSSS power level of 1657.1 MWt.

1.7 CONCLUSIONS

The results of evaluations/analyses described in the following sections lead to the following conclusions:

1. The Westinghouse fuel assemblies containing 14x14 422V+ upgraded fuel features for KNPP are mechanically compatible with the current Framatome/ANP fuel assemblies, control rods, flux detectors used in the instrumentation tubes, fuel handling equipment, fuel storage areas, and reactor internals interfaces. The current design bases for KNPP have been changed as described in this report to accommodate the 422V+ fuel assembly design.

2. The structural integrity of the 14x14 422V+ fuel assembly features has been evaluated for seismic/LOCA loadings for KNPP. Evaluation of the 422V+ fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the fuel assembly design is structurally acceptable with regard to grid crush. These faulted condition loads include seismic and LOCA forces.

3. Changes in the nuclear characteristics due to the transition to 14x14 422V+ fuel assembly features are addressed in Section 3.0 of the report. Changes in the nuclear characteristics in equilibrium cycles of the 422V+ fuel assembly will be within the range normally seen for Framatome/ANP fuel from cycle-to-cycle due to fuel management. The fuel management is specified by NMC in the Reload Schedule and Energy Requirements (RSER) document.

4. The reload 14x14 422V+ fuel assemblies are hydraulically compatible with the 14x14 Framatome/ANP fuel assemblies from previous reload cores.

5. The core design and safety analysis results documented in this report demonstrate the core's capability to operate safely with the parameter values that have been assumed for KNPP operation at uprated conditions.

6. This report establishes a reference upon which to base Westinghouse reload safety evaluations for future reloads with the upgraded fuel features and power level described herein.

Table 1-1 RTDP Uncertainties

Parameter	Calculated Uncertainty	Uncertainty Used in Safety Analysis
Power	±1.72% -0.32% bias	±2.0% -0.32% bias (at 1757 MWt-NSSS power)
Reactor Coolant System Flow	±2.86% +0.11% bias	±4.3% +0.11% bias
Pressure	±35.1 psi 15.0 psi bias	±50.0 psi 15.0 psi bias
Inlet Temperature	±4.9°F -1.1°F (bias)	±6.0°F -1.1°F (bias)

Table 1-2 Performance Capability Parameters

OWNER UTILITY:	Wisconsin Public Service Corp		PCWG-2534
PLANT NAME:	Kewaunee (WPS)		CATEGORY III - RCS
UNIT NUMBER:			
BASIC COMPONENTS			
Reactor Vessel, ID, in.	132	Isolation Valves	No
Core		Number of Loops	2
Number of Assemblies	121	Steam Generator	
Rod Array	Framatome[1]	Model	54F[4]
Rod OD, in.	(1)	Shell Design Pressure, psia	1100
Number of Grids	(1)	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	29	Pump Motor, hp	6000
Internals Type	RGE	Frequency, Hz	60

CURRENT DESIGN BASIS [7]

	Replacement Steam Generators and Modified NSSS Power[2,4]			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	100	100	100	100
MWt	1657.1 [2]	1657.1 [2]	1657.1 [2]	1657.1 [2]
10^6 BTU/hr	5654	5654	5654	5654
Reactor Power, MWt	1650	1650	1650	1650
10^6 BTU/hr	5630	5630	5630	5630
Thermal Design Flow, Loop gpm	89,000	89,000	89,000	89,000
Reactor 10^6 lb/hr	69.3	69.3	67.5	67.5
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.0[3]	7.0[3]	7.0[3]	7.0[3]
Reactor Coolant Temperature, °F				
Core Outlet	590.8	590.8	611.0	611.0
Vessel Outlet	586.3	586.3	606.8	606.8
Core Average	557.6	557.6	579.0	579.0
Vessel Average	554.1	554.1	575.3	575.3
Vessel/Core Inlet	521.9[6]	521.9[6]	543.8	543.8
Steam Generator Outlet	521.6	521.6	543.6	543.6
Steam Generator				
Steam Temperature, °F	497.3	493.8	520.3	517.0
Steam Pressure, psia	664[5]	644[5]	815[5]	791[5]
Steam Flow, 10^6 lb/hr total	7.11	7.11	7.14	7.14
Feed Temperature, °F	427.5	427.5	427.5	427.5
Moisture, % max.	0.25	0.25	0.25	0.25
Design F'F', hr. sq. ft. °F/BTU	0.00011	0.00011	0.00011	0.00011
Tube Plugging, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Pump Design Point, Flow (gpm)/Head (ft.)			89,000/259	
Mechanical Design Flow, gpm			101,100	

Notes:
1. Competitor fuel, PCWG parameters support operation with 14x14 W.
2. 2.1 MWt increase in NSSS power per customer request (NSD-SAE-ESI-00-121).
3. Value accounts for thimble plug deletion and bounds the 6.92 percent value specified by Wisconsin Public Service.
4. Parameters reflect Model 54F replacement steam generators.
5. 19 psi steam generator internal pressure drop incorporated.
6. Operating T_{cold} must be maintained at 525°F or greater.

Table 1-2 Performance Capability Parameters (cont.)

OWNER UTILITY:	Wisconsin Public Service Corp	PCWG-2707	
PLANT NAME:	Kewaunee (WPS)	CATEGORY III - RCS	
UNIT NUMBER:			
BASIC COMPONENTS			
Reactor Vessel, ID, in.	132	Isolation Valves	No
Core		Number of Loops	2
Number of Assemblies	121	Steam Generator	
Rod Array	14x14 422V+	Model	54F[2]
Rod OD, in.	0.422	Shell Design Pressure, psia	1100
Number of Grids	2I/5Z	Reactor Coolant Pump	
Active Fuel Length, in.	143.25	Model/Weir	93A/No
Number of Control Rods, FL	29	Pump Motor, hp	6000
Internals Type	RGE	Frequency, Hz	60

CURRENT DESIGN BASIS[7]

	Current Design Basis[4]	RTSR/Uprate Program[2]			
THERMAL DESIGN PARAMETERS	See previous sheet	Case 1	Case 2	Case 3	Case 4
NSSS Power, %		107.4	107.4	107.4	107.4
MWt		1780	1780	1780	1780
10^6 BTU/hr		6,074	6,074	6,074	6,074
Reactor Power, MWt		1,772	1,772	1,772	1,772
10^6 BTU/hr		6,046	6,046	6,046	6,046
Thermal Design Flow, Loop gpm		89,000[4]	89,000[4]	89,000[4]	89,000[5]
Reactor 10^6 lb/hr		69.34	69.34	67.87	67.87
Reactor Coolant Pressure, psia		2250	2250	2250	2250
Core Bypass, %		7.0[3]	7.0[3]	7.0[3]	7.0[3]
Reactor Coolant Temperature, °F					
Core Outlet		595.5	595.5	611.3	611.3
Vessel Outlet		590.8	590.8	606.8	606.8
Core Average		560.2	560.2	577.1	577.1
Vessel Average		556.3	556.3	573.0	573.0
Vessel/Core Inlet		521.9	521.9	539.2	539.2
Steam Generator Outlet		521.6	521.6	538.9	538.9
Steam Generator					
Steam Temperature, °F		495.9	492.1	514.0	510.4
Steam Pressure, psia		656[1]	634[1]	771[1,5]	747[1]
Steam Flow, 10^6 lb/hr total		7.74	7.73	7.76	7.76
Feed Temperature, °F		437.1	437.1	437.1	437.1
Moisture, % max.		0.25	0.25	0.25	0.25
Design F'F', hr. sq. ft. °F/BTU		0.00011	0.00011	0.00011	0.00011
Steam Generator Tube Plugging, %		0	10	0	10
Zero Load Temperature, °F		547	547	547	547
HYDRAULIC DESIGN PARAMETERS					
Pump Design Point, Flow (gpm)/Head (ft.)				89,000/259	
Mechanical Design Flow, gpm				102,800	
Minimum Measured Flow, gpm total				186,000[6]	

Notes:
1. Values include a 19 psi pressure drop
2. Parameters reflect Model 54F replacement steam generators
3. Values account for thimble plug deletion and bounds the 6.92 percent value specified by Wisconsin Public Service.
4. Maintained per customer request
5. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 809 psia, steam temperature of 519.4°F, and steam flow of 7.77×10^6 lb/hr total should be assumed.
6. Customer-specified value.

2 MECHANICAL DESIGN FEATURES

2.1 INTRODUCTION AND SUMMARY

This section evaluates the mechanical design of the 14x14 422V+ fuel design and its compatibility with the currently used 14x14 Framatome/ANP fuel assembly design. The evaluation covers the transition through mixed-fuel type core populations to cores with only 422V+ type fuel. The 422V+ fuel assembly has been designed to be compatible with the 14x14 Framatome/ANP fuel assembly, reactor internals interfaces, fuel handling equipment, and refueling equipment. The 422V+ design dimensions are essentially equivalent to the current KNPP 14x14 Framatome/ANP assembly design from an exterior assembly envelope and reactor internals interface standpoint (refer to Table 2-1). References in this section are made to WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report" (Reference 17), and WCAP-9500-A, "Reference Core Report 17x17 Optimized Fuel Assembly" (Reference 13).

The significant mechanical features of the 422V+ design, that are VANTAGE + features include the following:

- ZIRLO™ clad fuel rod,
- ZIRLO guide thimble and instrumentation tubes,
- Annular or solid half-inch pellets in top and bottom axial blankets
- Modified fuel assembly dimensions for high burnup design
- Fuel rod design modifications for high burnup design.

The significant mechanical features of the 422V+ design, that are PERFORMANCE + features include the following:

- Low cobalt top and bottom nozzles
- Zirconium oxide coated lower fuel rod (optional)
- ZIRLO™ mid-grids, and
- Mid-enrichment of the annular or solid pellets in axial blankets

Note: Oxide coating is not available on gadolinium burnable absorber rods.

Other significant mechanical features of the 422V+ design include the following:

- 0.422-inch outer diameter (OD) fuel rod

- 0.422-inch OD instrumentation tube

- New OFA style (2.25-inches tall, vertical springs, horizontal dimples) mid-grid has been designed to be compatible with the 0.422-inch OD fuel rods. The Mechanical Compatibility Report (Reference 64) and Section 2.2 confirms that the new mid-grid design on the 422V+ fuel is compatible with the Framatome/ANP fuel currently in the KNPP core.

The ZIRLO™ 422V+ mid-grid design combines enhanced anti-snag geometry and reduced pressure drop performance. This mid-grid design evolution started with the original STANDARD Inconel mid-grid,

which was used in KNPP prior to transitioning to Framatome STD fuel. This design was modified to an OFA style zircaloy vaneless design that has been used extensively in the Zorita Nuclear Power Plant and in two regions in Point Beach. From the Zorita design, the mid-grid was adapted for Point Beach and KNPP by adding mixing vanes, changing the material composition from Zircaloy-4 to ZIRLO, and other minor design changes. Additional discussions of this mid-grid design are covered in Section 2.2.2.

Section 2.2 describes and evaluates the differences between the fuel assembly designs.

Based on evaluation of the 422V+ and the 14x14 Framatome/ANP design differences, it is concluded that the two designs are mechanically compatible with each other. The 422V+ fuel rod mechanical design bases are the same as those used for the optimized fuel assembly (OFA) assemblies in the previously supplied Lead Use Assemblies (LUAs), currently operating in KNPP cycle 25, and are similar to the 422V+ fuel assemblies in their second cycle of operations in Point Beach Units 1 and 2.

2.2 COMPATIBILITY OF FUEL ASSEMBLIES

Table 2-1 compares the 14x14 Framatome/ANP and 422V+ design parameters for KNPP. Figure 2-1 depicts the 14x14 Framatome/ANP and 422V+ assembly designs noting respective overall height and grid elevation dimensions. The Westinghouse OFA top and bottom nozzles are used in the 422V+ fuel assemblies, except that they are made of low cobalt material. The 422V+ assembly skeleton is similar to that previously described for OFA (References 13 and 18), except for those modifications necessary to accommodate intended fuel operation to higher burnup levels and the new mid-grid design and instrumentation tube sizing. The OFA top and bottom nozzles will be used in the 422V+ fuel assembly, except that they will be made of low cobalt material. The other modifications consist of the use of ZIRLO™ guide thimble tubes and small skeleton dimensional alterations to provide additional fuel assembly and rod growth space at extended burnup. The 422V+ fuel assembly is 0.065 inches longer than the 14x14 Framatome/ANP design. The structural mid-grid centerline elevation of the 422V+ assembly is 0.255 inches higher than that of the 14x14 Framatome/ANP assembly. Variations in the top-grid and bottom-grid centerline elevations relative to the 14x14 Framatome/ANP design are 0.425 inches and 0.113 inches, respectively. This variation, with the 422V+ top-grid centerline being lower than that of the 14x14 Framatome/ANP assembly, accounts for the difference in growth rate of Zircaloy-4 and ZIRLO™ and accommodates the higher burnup capability of the 422V+ design. By designing a grid offset in the fuel and accounting for growth variations between Zircaloy-4 and ZIRLO™ , the end-of-life (EOL) grid elevations will maintain sufficient overlap.

Crossflows can result from a mismatch in axial location of grids in adjacent assemblies. These mismatches can be a result of design differences or irradiation growth of the fuel assemblies. Crossflows can also result from a mismatch in grid loss coefficients if different grid types are used in adjacent assemblies. Allowable grid mismatches will occur for physically identical fuel assemblies in reload regions due to axial mismatches caused by assembly tolerances and irradiation growth of the fuel assemblies. Allowable grid mismatches can also result for fuel assemblies having identical grid loss coefficients but with the fuel assemblies being intentionally designed with a slight amount of grid offset. The criterion is that there should be outer grid strap overlap between any two fuel assemblies in the core throughout their life in the core. For the 422V+ fuel assembly design, it has been determined that outer grid strap overlap will be maintained throughout the assembly's residency in the core when next to a

14x14 Framatome/ANP assembly. The 14x14 Framatome/ANP assembly design uses a Zircaloy-4 skeleton whereas the 422V+ fuel assembly design uses a ZIRLO skeleton. Since ZIRLO growth is less than Zircaloy-4 growth, this has been accounted for in determining grid mismatches between the two assembly designs to ensure that sufficient overlap remains throughout the fuel assemblies' residency in the core.

Since the 14x14 422V+ fuel is intended to replace either the Westinghouse 14 x 14 STD, or the 14 x 14 OFA fuel assembly designs, the 422V+ exterior assembly envelope is equivalent in design dimensions. The functional interface with the reactor internals is also equivalent to those of previous Westinghouse fuel designs. The 422V+ fuel assembly is designed to be mechanically and hydraulically compatible with the STANDARD and OFA in full or transition cores. The NRC has reviewed and approved the use of 422V+ fuel in the Point Beach Units 1 and 2. The 14x14 STD fuel has been used in the KNPP plant prior to transitioning to the SPC (Framatome/ANP) fuel.

2.2.1 Fuel Rods

The 422V+ fuel rod has a clad thickness comparable to the 14x14 Framatome/ANP fuel rod. The 422V+ cladding is ZIRLO instead of Zircaloy-4 to enhance fuel reliability. The 422V+ rods are longer to provide additional plenum volume to accommodate fission gas release at extended burnup. The 422V+ rods also have a shorter pellet stack height and annular blanket pellets to provide additional volume. The bottom end-plug has an internal grip feature to facilitate rod loading and to provide appropriate lead-in for the removable top nozzle reconstitution feature. The KNPP 422V+ fuel rod also may have a zirconium oxide coating at the bottom end of the fuel rod to provide additional rod fretting wear protection early in life before the natural oxide layer builds up during in-reactor operations. The 422V+ fuel also has mid-enriched solid or annular pellets in the axial blankets.

The 422V+ design, containing the mid-enriched annular pellets in axial blankets, will behave in a similar manner to the 14x14 Framatome/ANP fuel of comparable enrichments and burnups containing solid pellets in axial blankets with regard to xenon stability, load follow capability, peaking factors, rod worths and shutdown margin. The axial power distribution for fuel with annular axial blankets will be slightly more peaked than for fuel having comparable solid axial blanket pellets. This slight increase in the axial power shapes would be no more than that seen due to cycle-to-cycle variations in fuel management. Annular pellets provide additional plenum volume for fission gas releases (Reference 12) while axial blankets reduce neutron leakage and improve fuel utilization.

The key design difference between mid-enriched annular pellets in axial blankets and enriched solid fuel pellets—aside from the U-235 enrichment—is the annulus itself. The annulus volume is approximately 25 percent of the total pellet volume. Annular pellets in axial blankets have the same chamfer as the enriched solid fuel pellets, but no dish on the pellet ends. Pellet length-to-diameter ratio is maintained at approximately 1.4. This ratio has been adjusted to produce an even multiple of pellet lengths to obtain appropriate axial blanket zone lengths in fabrication. A reduction in the fuel stack length of 0.75 inches along with 6 inches of annular pellets in axial blankets will be used in the 422V+ fuel design. This stack reduction is incorporated to provide additional plenum volume space for gas releases.

The relatively low range of linear heat rate which the annular pellets in axial blankets will experience, and the modest fraction of the fuel volume which it occupies, assures that their use will not have any

significant effect on the limiting fuel temperature or rod internal pressure, other than that due to the additional void volume provided by the axial blanket pellet annulus (References 12 and 19).

A factor requiring consideration when evaluating the pellet-to-cladding gap is the cladding material. The 422V+ fuel uses ZIRLO™ cladding instead of Zircaloy-4 as does the 14x14 Framatome/ANP fuel. The resultant effect on pellet-to-cladding contact due to the different creep rates of ZIRLO and Zircaloy-4 is negligible.

The 422V+ fuel rod design bases and evaluation are given in Section 2.0 of Reference 17.

2.2.2 Grid Assemblies

The top and bottom Inconel (non-mixing vane) grids of the 422V+ fuel assemblies are similar in design to the current Inconel grids used with OFA fuel assemblies, except for the cell sizing changes required to accommodate the larger diameter fuel rod and instrumentation tube. The ZIRLO interlocking strap and grid/sleeve joints are laser welded, whereas the Inconel grid joints are brazed.

The other significant change between the two designs, other than ZIRLO, is the development of a LPD mid-grid. This mid-grid design evolution started with the original STANDARD Inconel mid-grid. The design was modified to an OFA style zircaloy vaneless design that has been used extensively in the Zorita Nuclear Power Plant. From the Zorita design, the mid-grid was adapted to the Point Beach and KNPP design by adding mixing vanes, changing the material composition from Zircaloy-4 to ZIRLO, and other minor design changes. This new mid-grid design has been developed and licensed under the guidelines of the Fuel Criteria Evaluation Process (FCEP) (Reference 20). By complying with the requirements of FCEP, it has been demonstrated that the new mid-grid meets all design criteria of existing tested mid-grids that form the basis of the WRB-1 correlation database and that the WRB-1 correlation with a 95/95 correlation limit of 1.17 applies to the new mid-grid. This FCEP applicability demonstration was presented to the NRC in a meeting held on August 5, 1997. More details of this presentation are documented in the slide presentation that has been forwarded to the NRC for the record(Reference21).

The 422V+ grid assembly design bases and evaluation are given in Section 2.3 of Reference 17.

2.2.3 Guide Thimble and Instrumentation Tubes

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 (ID) provides a minimum diametral clearance of 0.0088 inches (under worst case conditions) for control rods supplied by Westinghouse. The 422V+ instrumentation tube diameter is 0.424 inches, compared to the 14x14 Framatome/ANP design value of 0.424 inch. The other significant difference between the 422V+ and 14x14 Framatome/ANP design is the change from Zircaloy-4 to ZIRLO material.

The general design bases for the 422V+ guide thimble and instrumentation tubes remain the same as those given in Reference 1.

2.2.4 Reconstitutable Top Nozzle (RTN)

The 422V+ RTN will be fabricated of low cobalt 304 stainless steel to reduce as-low-as-reasonably-achievable (ALARA) concerns. The 422V+ RTN, which is similar to the Westinghouse 14x14 OFA RTN, is in its second cycle of operation in Point Beach Units 1 and 2. The change in material does not impact any design criterion.

2.2.5 Debris Filter Bottom Nozzle (DFBN)

The bottom nozzle will also be fabricated of low cobalt 304 stainless steel to reduce ALARA concerns. The change in material does not impact any design criterion. The 422V+ debris filter bottom nozzle (DFBN) is identical to the 14x14 Framatome/ANP bottom nozzle with respect to the envelope (7.761 inches). The 422V+ DFBN has a height 0.217 inches less than the 14x14 Framatome/ANP bottom nozzle.

2.3 MECHANICAL PERFORMANCE

Design changes associated with the 14x14 422V+ design do not significantly influence the fuel assembly structural characteristics that were determined by prior mechanical testing (see Reference 64). Therefore, the 422V+ fuel assembly, with expected structural behavior and projected performance, will meet design requirements throughout the fuel's life.

2.4 FUEL ROD PERFORMANCE

Fuel rod performance for 422V+ KNPP fuel is identical to 422V+ fuel in use at the Point Beach Units 1 and 2, which has previously been shown to satisfy the NRC Standard Review Plan (SRP) (Reference 21) fuel rod design bases on a region-by-region basis. The design bases for Westinghouse 422V+ fuel are discussed in Reference 17.

There is no impact from a fuel rod design standpoint of having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for 422V+ fuel incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions (such as the presence of annular pellets in axial blankets or changes in the fuel rod diameter and plenum length). Fuel performance evaluations have been completed for 422V+ fuel to demonstrate that the design criteria will be satisfied in the core under the planned operating conditions of a power uprating to 1772 MWt core power and the peaking factor $F^N_{\Delta H}$ limit of 1.70. Any additional changes from the plant operating conditions originally evaluated for the mechanical design will be addressed for all affected 422V+ fuel regions as part of the cycle-specific reload safety evaluation process when the plant changes are to be implemented.

Fuel rod design evaluations for the 422V+ fuel were performed using NRC-approved models (References 17, 23, and 44) and NRC-approved design criteria methods (References 24 and 25) to demonstrate that all fuel rod design criteria are satisfied. Approval of Reference 68 by the NRC and subsequent re-evaluation of the stress values will be conducted to confirm the new clad stress criteria is met.

The fuel rod design criteria given below are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod having the minimum margin to the design limit. In general no single rod is limiting with respect to all the design criteria. Generic evaluations have identified which rods are most likely to be limiting for each criterion, and exhaustive screening of fuel rod power histories to determine the limiting rod is typically not required.

The NRC-approved Performance, Analysis, and Design (PAD) 3.4 and PAD 4.0 codes, with NRC-approved models (References 23 and 44) for in-reactor behavior, is used to calc ulate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD interactively calculates the interrelated effects of temperature, pressure, clad elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

PAD 3.4 and PAD 4.0 are best estimate fuel rod performance models, and in most cases the design criterion evaluations are based on a best-estimate-plus-uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties are measured for some critical inputs, such as, fuel pellet diameter. When available, measured parameters can be used in lieu of fabrication uncertainties.

The COROSN code is used to evaluate clad and structural component oxidation and hydriding. COROSN uses the same thermal, corrosion and hydriding models as PAD and is specially adapted for efficient evaluation of the oxidation and hydriding design criteria.

2.4.1 Fuel Rod Design Criteria

The criteria pertinent to the fuel rod design are as follows:

- Rod Internal Pressure
- Clad Stress and Strain
- Clad Oxidation and Hydriding
- Fuel Temperature
- Clad Fatigue
- Clad Flattening
- Fuel Rod Axial Growth
- Plenum Clad Support
- Clad Free-Standing
- End-plug Weld Integrity

The specific assumptions used in the verification of these criteria for KNPP fuel include:

- KNPP Fuel Upgrade/Power Uprating (FU/PU) specific operating conditions
- Fuel rod duty (steady state powers, fuel rod axial power shapes, etc.)

Each of these key fuel rod design criteria has been evaluated for use of the Westinghouse 422V+ fuel assembly design in KNPP. Based on these evaluations, it is concluded that each design criterion can be satisfied through transition cycles to a full core of the 422V+ design. Approval of Reference 68 by the NRCand subsequent re-evaluation of the stress values are necessary to confirm that the proposed clad stress crtieria is met. The design criteria are described in more detail below.

Rod Internal Pressure

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which could cause:

- The diametral gap to increase due to outward clad creep during steady state operation
- Extensive DNB propagation to occur.

The rod internal pressure for the KNPP 422V+ fuel rods has been evaluated by modeling the gas inventories, gas temperature and rod internal volumes through the rods' life. The resulting rod internal pressure is compared to the design limit on a case-by-case basis of current operating conditions to EOL. This evaluation showed that the rod internal pressure satisfies the design limit.

The second part of the rod internal pressure design basis precludes extensive DNB propagation and associated fuel failure. The basis for this criterion is that no significant additional fuel failures due to DNB propagation will occur in cores which have fuel rods operating with rod internal pressure in excess of system pressure. 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. For Condition III/IV events, it is shown that the total number of rods in DNB, including propagation effects, is consistent with the assumptions used in radiological dose calculations for the event under consideration. For the KNPP 422V+ FU/PU program, Condition III/IV analysis assumes a wide range of number of rods in DNB (from 5 percent to 40 percent of the rods) to cover all potential situations.

Clad Stress and Strain

The design limit for clad stress is that the volume average effective stress considering (a) interference due to uniform cylindrical pellet-to-clad contact caused by pellet thermal expansion, pellet swelling and uniform clad creep, and (b) pressure differences between the rod internal pressure and the system coolant pressure is less than the clad yield strength for Condition I and II events. While the clad has some capability for accommodating plastic strain, the yield stress has been established as the conservative design limit. The design limit for clad strain during steady-state operation is that the total plastic tensile creep strain due to uniform clad creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1 percent from the unirradiated condition. The design limit for fuel rod clad strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1 percent from the pre-transient value. These limits are consistent with proven practice.

While the strain criteria can be met, the cladding stress criterion is violated. Westinghouse submitted in Reference 68 (Addendum 2 to WCAP-12488) a revised cladding stress criterion consistent with current industry guidelines to the NRC. Once approved, the design limit for cladding stress would become that the maximum cladding stress intensities excluding pellet cladding interaction but accounting for cladding corrosion as a loss-of-load carrying metal, be less than the stress limit, based on the ASME code calculations.

Pending approval of the proposed methodology, the stress values will be reevaluated to confirm the new stress design limits are met.

Clad Oxidation and Hydriding

The design criteria related to clad corrosion require that the Zircaloy-4/ZIRLO clad metal-oxide interface temperature be maintained below specified limits to prevent a condition of accelerated oxidation which would lead to clad failure. The calculated clad temperature (metal-oxide interface temperature) will be less than 750°F/780°F during steady-state operation and for Condition II transients, the calculated clad temperature will not exceed 800°F/850°F, respectively for Zircaloy-4 and ZIRLO clad. The clad surface temperatures were evaluated and satisfied the above temperature limits. The base metal wastage of the Zircaloy-4 and ZIRLO™ grids and guide tubes was shown not to exceed a 12-percent design limit at EOL.

The best estimate hydrogen pickup level in Zircaloy-4/ZIRLO clad and Zircaloy-4/ZIRLO structural components shall be less than or equal to 600 ppm on a volumetric average basis at EOL. The hydrogen pickup criterion, which limits the loss of ductility due to hydrogen embrittlement which occurs upon the formation of zirconium hydride platelets, has been met with the current approved model for the KNPP FU/PU Program.

Fuel Temperature

For Condition I and II events, the fuel and reactor protection system are designed to assure that a calculated centerline fuel temperature does not exceed the fuel melting temperature criterion. The intent of this criterion is to avoid a condition of gross fuel melting which can result in severe duty on the clad. The concern here is based on the large volume increase associated with the phase change in the fuel and the potential for loss of clad integrity as a result of molten fuel/clad interaction.

The temperature of the fuel pellets was evaluated by modeling the fuel rod geometry, thermal properties, heat fluxes and temperature differences in order to calculate fuel surface, average and centerline temperatures of the fuel pellets.

Fuel temperatures have been calculated as a function of local power and burnup. The fuel surface and average temperatures with associated rod internal pressure are provided to Transient Analysis and LOCA for accident analysis of the 422V+ fuel design (see Section 4.5 for additional information). The fuel centerline temperature is used to show that fuel melt will not occur. For 422V+, the local linear power which precludes fuel centerline melting is 22.54 kW/ft.

Clad Fatigue

The fuel rod design criterion for clad fatigue requires that, for a given strain range, the number of strain fatigue cycles is less than those required for failure considering a factor of safety of 2.0 on the stress amplitude and a factor of safety of 20.0 on the number of cycles. This criterion addresses the accumulated effect of short-term, cyclic clad stress and strain which results from daily load follow operation.

Clad fatigue for the KNPP 422V+ fuel was evaluated by using a limiting fatigue duty cycle consisting of daily load follow maneuvers. The evaluation showed that the cumulative fatigue usage factor is less than the design limit of 1.0.

Clad Flattening

The clad flattening criterion prevents fuel rod failures due to long-term creep collapse of the fuel rod clad into axial gaps formed within the fuel stack. Current fuel rod designs employing fuel with improved in-pile stability provide adequate assurance that axial gaps large enough to allow clad flattening will not form within the fuel stack.

The NRC has approved WCAP-13589-A (Reference 25), which provided data to confirm that significant axial gaps in the fuel column due to densification (and therefore clad flattening) will not occur in current Westinghouse fuel designs. The KNPP 422V+ fuel meets the criteria for applying the Reference 25 methodology and, therefore, clad flattening will not occur.

Fuel Rod Axial Growth

This criterion assures that sufficient axial space exists to accommodate the maximum expected fuel rod growth without degradation of the assembly function. Fuel rods are designed with adequate clearance between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly to preclude interference between these members.

The KNPP fuel rod growth evaluation demonstrates that there is adequate margin to the fuel rod growth design limit for the 422V+ fuel.

Plenum Clad Support

This criterion assures that the fuel clad in the plenum region of the fuel rod will not collapse during normal operating conditions, or distort so as to degrade fuel rod performance. The helical coil spring used in the 422V+ fuel design prevents potential clad collapse by providing clad support.

Clad Free Standing

The clad free-standing criterion requires that the clad shall be short-term free standing at beginning of life, at power, and during hot hydrostatic testing. This criterion precludes the instantaneous collapse of the clad onto the fuel pellet caused by the pressure differential across the clad wall.

Evaluations of the clad-free-standing criterion have shown that instantaneous collapse of the KNPP fuel will be precluded for differential pressures well in excess of the maximum expected differential pressure across the clad under operating conditions.

Fuel rod design evaluations for KNPP are performed using the NRC approved models in References 17 and 23 to demonstrate that the SRP fuel rod design criteria are satisfied.

End-plug Weld Integrity

The fuel rod end-plug weld shall maintain its integrity during Condition I and II events and shall not contribute to any additional fuel failures above those already considered for Condition III and IV events. The intent of this criterion is to ensure that fuel rod failures will not occur due to tensile pressure differential loads which can exist across the weld. The current inspection limits for the end-plug weld allow for the existence of small defects within the weld, and under maximum tensile pressure differential, failure of the weld shall not occur.

For Condition I and II events, the methodology is to confirm that the cold and hot internal pressure values of 1400 psia (70°F) and 3500 psia, respectively, are bounding for the fuel regions in each reload cycle. For Condition III and IV events, the weld plug integrity methodology determines the maximum tensile pressure differential load during the return-to-power phase of the hot zero power (HZP) steam line break (SLB) event. This is done by evaluating the rod internal pressure during the transient of the limiting rod and subtracting the minimum system pressure during the transient to determine the maximum tensile pressure differential. The maximum tensile pressure differential load is compared to the allowable pressure differential load at the minimum transient temperature to determine if the weld integrity criteria are satisfied.

For the KNPP 422V+ fuel design, this criterion has been shown to be met.

2.4.2 Oxide-to-Metal Ratio

When water reacts with zirconium based alloys, the surface of the metal is converted to an oxide. Due to the differences in the densities of the oxide and the base metal, there is a volumetric change from the metal consumed to the oxide generated. This volumetric difference results in a thicker oxide than the metal that was consumed. The ratio of the volumes is characterized by the oxide-to-metal ratio (O/M). The theoretical oxide-to-metal ratio is referred to as the Pilling-Bedworth ratio. For zirconium based alloys, Westinghouse uses a value of 1.56. The O/M ratio is used in the fuel rod design for steady-state calculations to determine the maximum steady-state oxide thickness. It is also used by the LOCA analysis group to determine the maximum transient oxide thickness that would occur during a LOCA event. These two oxide thicknesses are added together to ensure that the total localized oxidation does not exceed the 10 CFR 50.46 criterion of 17 percent.

For KNPP 422V+ fuel, the steady-state oxidation will be considerably less than it is for the Zircaloy-4 14x14 Framatome/ANP fuel since the ZIRLO corrosion rate is ~75 percent of that for Zircaloy-4. Since the transient oxidation adder is relatively small in comparison to the steady-state oxidation component, the 10 CFR 50.46 oxidation criterion of 17 percent will not be challenged for the FU/PU program.

2.5 SEISMIC/LOCA IMPACT ON FUEL ASSEMBLIES

The 422V+ fuel assembly design is comparable to the 14x14 Framatome/ANP assembly. Seismic/LOCA analyses demonstrated adequate grid load margin on all fuel assemblies except for the fuel assemblies (FAs) on the periphery of the 13 assembly row in the limiting mixed Condition I. (See Figure 2-2.) Evaluations demonstrated that the core (except for the fuel assemblies noted) coolable geometry and control rod insertion requirements are met.

An evaluation of 422V+ and 14x14 Framatome/ANP assembly structural integrity, considering the lateral effects of LOCA auxiliary line breaks (accumulator lines) and two safe shutdown earthquake (SSE) seismic cases (Cases A and B) (Reference 69), has been performed. The SSE/LOCA analysis results were obtained using the time history numerical integration technique. The maximum grid impact forces obtained from both transients were combined using the square root of the sum of the squares (SRSS) method. The maximum loads were compared with the allowable grid crush strength. This analysis is discussed in more detail in the following paragraphs.

2.5.1 Fuel Assembly and Reactor Core Models

Based on the assembly vibration frequencies and mode shapes, a parametric study was performed using NKMODE. NKMODE calculates a set of equivalent spring-mass elements representing an individual fuel assembly structural system. Based on this model, it has been shown that the mode shapes agree well with the predominate fuel assembly vibration frequencies. The lumped mass-spring fuel assembly model was further verified using the WECAN finite element code.

With the appropriate analysis parameters such as grid impact stiffness and damping, number of fuel assemblies in a planar array and gap clearance established, the WEGAP reactor core model was used for analyzing transient loads.

2.5.2 Grid Load Analysis

The time history motions of the core barrel at the upper core plate elevation and the upper and lower core plates are applied simultaneously to the reactor core model. The time histories representing the SSE motion and the pipe rupture transients were obtained from the time history analyses of the reactor vessel and internals finite element model.

Homogeneous Core

The maximum structural grid loads for the fuel assemblies occurred in the peripheral assemblies in the 13 fuel assembly array. The maximum fuel assembly deflection was 0.7974 inches. The maximum grid loads obtained from the SSE and LOCA analyses were combined using the SRSS method. The results of the combined SSE and LOCA analyses indicate that the maximum impact forces for the 422V+ assembly design using the 2-directional grid characteristics are 85 percent and 91 percent of the respective allowable grid strengths. The allowable grid strengths are established at the 95-percent confidence level on the true mean from the distribution of experimentally determined grid crush data at operating temperature.

Mixed Cores

The maximum structural grid loads for the 422V+ assemblies occurred in the peripheral assemblies in the 13 fuel assembly array. The maximum fuel assembly deflection was 0.872 inches. The results of the combined seismic and LOCA analyses indicate that the maximum impact forces for the 422V+ assembly design using the 2 directional grid characteristics are less than the respective allowable grid strengths (except for the 13 fuel assembly row in mixed condition I). The maximum impact forces for the 14x14 Framatome/ANP assemblies are 87.5 percent of the respective allowable grid strengths. The allowable

grid strengths are established at the 95-percent confidence level on the true mean from the distribution of experimentally determined grid crush data at operating temperature.

Based on the results of the combined SSE and LOCA loads and the core coolable geometry assessment, the 422V+ fuel assembly and the Framatome/ANP fuel assembly designs are structurally acceptable for the KNPP homogeneous and transition cores.

Stress

Fuel assembly displacement is limited by the total accumulated gap clearances plus grid deformations. Fuel assembly stresses were calculated based on the most limiting case. Stresses for the fuel rods and thimble tubes were calculated based on a vertical impact load of 4000 lbs., a 1.0-inch fuel assembly lateral deflection, and operating condition loads. The results indicate that there are adequate margins for both fuel rods and thimble tubes, eliminating the possibility of fuel rod fragmentation. The reactor can be safely shut down under faulted condition loading. The 422V+ assembly design is structurally acceptable under combined seismic and LOCA loads for KNPP.

2.5.3 Conclusions

The maximum horizontal input motion congruent with the core principal axis is used to determine dynamic fuel responses. The reactor core is analyzed as a de-coupled system with respect to the two lateral directions. The input forcing function is obtained from a separate reactor pressure vessel and reactor internals system analysis.

The evaluation of the 422V+ fuel assembly in homogeneous cores in accordance with NRC requirements as given in SRP 4.2, Appendix A (Reference 22), shows that the 422V+ fuel is structurally acceptable for the KNPP reactor. The grid loads evaluated for the LOCA and seismic events and combined by the SRSS method identified in SRP 4.2 are less than the allowable limit. The same conclusion is true for a transition core composed of both 422V+ fuel assemblies and 14x14 Framatome/ANP assemblies, except for the 13 fuel assembly row in mixed condition I. An additional evaluation demonstrated that the core coolable geometry is maintained. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are within acceptable limits for the 422V+ fuel. The reactor can be safely shutdown under combined faulted condition loads.

2.6 CORE COMPONENTS

The core components for the KNPP are evaluated to be compatible with the 422V+ fuel design. The 422V+ guide thimble tubes provide sufficient clearance for insertion of control rods. KNPP has also been designed to operate with thimble plugs removed.

Table 2-1 Comparison of 14x14 Framatome/ANP, and Westinghouse 422V+ Fuel Assembly Mechanical Design Parameters

	Framatome/ANP	422V+
Fuel Assembly Overall Length, inches	159.70	159.775
Fuel Rod Overall Length, inches	152.07	152.563
Nominal Assembly Envelope at Bottom Nozzle, inches	7.761	7.761
Fuel Rod Pitch, inches	0.556	0.556
Number of Fuel Rods/Assembly	179	179
Number of Guide Thimbles/Assembly	16	16
Number of Instrumentation Tubes/Assembly	1	1
Fuel Tube Material	Zircaloy-4	ZIRLO™ (coated bottom)
Fuel Tube Cladding OD, inches	0.424	0.422
Fuel Rod Cladding Thickness, inches	0.025	0.0243
Fuel Clad Gap, mil	3.50	3.75 (uncoated pellets)
Enriched Fuel Pellet diameter, inches	0.3670	0.3659 (uncoated pellets)
Enriched Fuel Pellet length, inches	0.440	0.4390
Annular Axial Blanket Pellet diameters		
ID, inches	N/A	0.1830
OD, inches	N/A	0.3659
Annular Axial Blanket Pellet length, inches	N/A	0.5450
Fuel Rod End Plugs	N/A	Tapered and Radiused (coated w/fuel rod)
Fuel Stack Height (cold, undensified), inches	144	143.25
Annular Axial Blanket Length		
(top and bottom), inches	N/A	6
Plenum volume, $inch^3$	N/A	0.8526
Guide Thimble Material	Zircaloy-4	ZIRLO™
Guide Thimble OD, inches	0.541	0.526
Guide Thimble Wall Thickness, inches	0.017	0.017

Table 2-1 (cont.) Comparison of 14x14 Framatome/ANP, and Westinghouse 422V+ Fuel Assembly Mechanical Design Parameters		
	Framatome/ANP	**422V+**
Grid Material, Inner		
Mid-grid (5 per Fuel Assembly)	Zircaloy-4	ZIRLO™
Edges Modified	N/A	Yes
Grid Material, End		
Grids (2 per Fuel Assembly)	Bimetallic, Zircaloy-4/ Inconel Spring	Inconel
Inner Spring (mid-grids)	Vertical	Vertical
Grid Fabrication		
Inconel Grids	Brazed Joining of Interlocking Stamped Straps	Brazed Joining of Interlocking Stamped Straps
Zircaloy-4 (mid-grids)	N/A	None
ZIRLO™ (mid-grids)	N/A	Laser Weld Joining of Interlocking Stamped Straps
Grid/Guide Thimble Attachment		
Inconel Grids	N/A	Thimbles Bulged Together with Sleeve Prebrazed
Zircaloy-4/ZIRLO™ (mid-grids)	N/A	Thimbles Bulged Together with Sleeves Laser Prewelded to Grid Straps
Relative Clad Thickness/Diameter Ratio	1.00	0.977
H_2O/UO_2 Volume Ratio (cold)	1.59	1.61
Relative UO_2/Rod	1.0	0.971
Top Nozzle	N/A	Reconstitutable Stainless Steel, Reduced Height, Removable Design, Low Cobalt
Compatible with Fuel	Yes	Yes

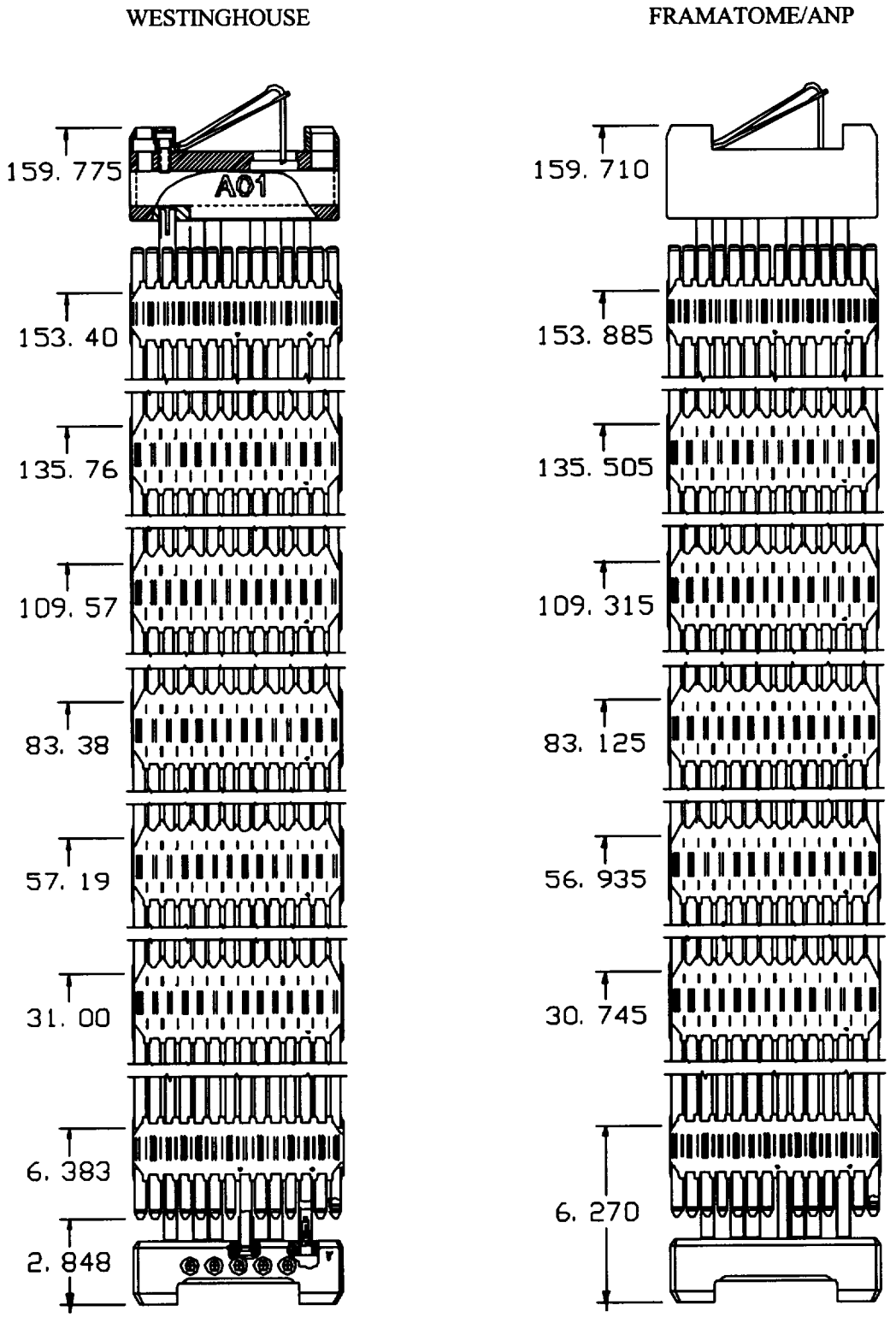

Figure 2-1 Westinghouse 422V+ versus Framatome Fuel Assembly Designs (Units in inches)

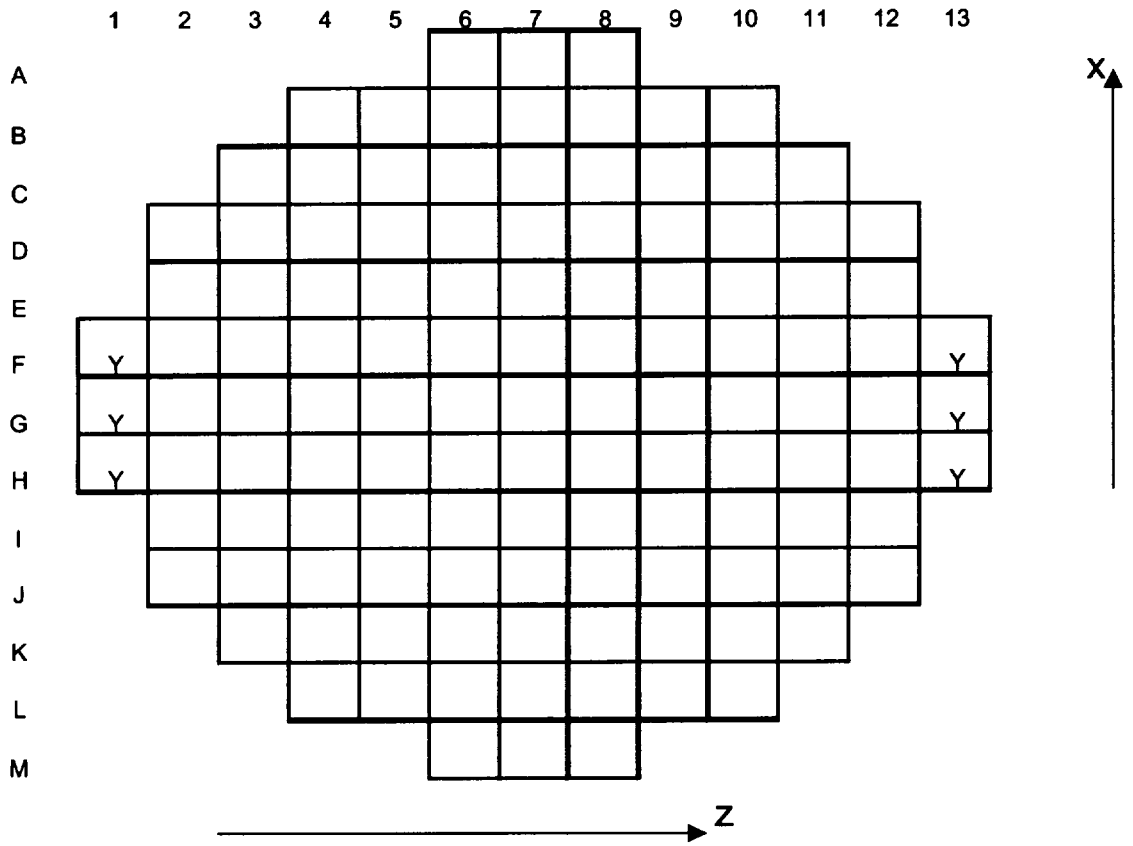

Figure 2-2 Locations of the Fuel Assembly with the Crushed Grid for KNPP Mixed Core

3 NUCLEAR DESIGN

3.1 INTRODUCTION AND SUMMARY

The effects of 1) using the Westinghouse VANTAGE+ fuel and fuel with PERFORMANCE+ features (422V+), 2) using both plant non-uprated and uprated conditions, and 3) using 18-month cycles on the nuclear design bases and methodologies for Kewaunee Nuclear Power Plant (KNPP) are evaluated in this section.

The specific values of core safety parameters, (such as, power distributions, peaking factors, rod worths, and reactivity parameters) are primarily loading pattern dependent. The variations in the loading-pattern-dependent safety parameters are expected to be typical of the normal cycle-to-cycle variations for standard fuel reloads. Margin to key safety parameter limits is not reduced by the 422V+ fuel design relative to the 14x14 Framatome/ANP design in similar applications. Standard nuclear design analytical models and methods (Reference 2, 27 and 28) accurately describe the neutronic behavior of the 422V+ fuel design.

Storage of the 422V+ upgraded fuel at KNPP was reviewed with respect to criticality effects, heat transfer capability of the spent fuel pool cooling system, gamma heating effects, and structural loading. Spent fuel criticality analyses for the 422V+ fuel were performed by Westinghouse (Reference 65). The analyses concluded that the spent fuel pool k_{eff} remains below the 0.95 limit for 422V+ fuel. Westinghouse also performed criticality calculations for the new fuel vault that demonstrate that k_{eff} remains below the 0.95 limit for new fuel assemblies having the maximum loading of 56.31 grams U235 if fully flooded with water. The new fuel vault k_{eff} remains below 0.98 for the fully flooded condition, and 0.98 if moderated by aqueous form. Therefore, storage of 422V+ fuel meets the criteria for spent and new fuel storage.

The plant Technical Specifications were reviewed and marked-up changes are included in Appendix A. The Technical Specification changes which impact the nuclear design are at uprated power conditions and uprated trip setpoints. Even though the nuclear design analyses were performed at uprated power conditions, the power uprate change is currently being evaluated by Nuclear Management Company (NMC) and will be submitted appropriately following the 422V+ Fuel Upgrade (FU) Program.

In summary, the changes from the current Framatome/ANP fuel core to a core containing the upgraded fuel product will not cause changes to the current KNPP USAR (Reference 1) nuclear design bases.

3.2 DESIGN BASIS

The specific design bases and their relation to the General Design Criteria (GDC) in 10 CFR 50, Appendix A for the 422V+ design are the same as those of the optimized fuel assembly (OFA) design (Section 3.1 of Reference 17). For the 422V+ product, the fuel burnup design is extended to a lead rod burnup of up to 75,000 MWD/MTU (Note: VANTAGE + is currently licensed to 60,000 MWD/MTU by the NRC (Reference 17) with extension to 62,000 MWD/MTU on a cycle specific basis, as delineated in Reference 20, Appendix R).

The 422V+ fuel design differs from that of the 14x14 Framatome/ANP design with the unique features as described in Section 2 of this report. Two features in the 422V+ design that are not present in the 14x14 Framatome/ANP design, which affect nuclear design, are: 1) use of ZIRLO material for fuel cladding, guide thimble tubes, instrumentation tubes, and LPD mid-grids, and 2) a changed fuel stack height within the assembly. The only substantial effect of the ZIRLO alloy on the nuclear design is attributable to an increase in fuel management flexibility provided by increased lead rod burnups. The 422V+ fuel assembly will have a fuel stack height reduction of 0.75 inches to accommodate fission gas release from the extended burnups of the 422V+ design.

The effects of extended burnup on nuclear design parameters have been previously discussed in Reference 24. That discussion is valid for the anticipated 422V+ design discharge burnup. In accordance with the NRC recommendation made in their review of Reference 29, Westinghouse will continue to monitor predicted versus measured physics parameters for extended burnup applications.

3.3 METHODOLOGY

The purpose of this reload transition core analysis is to determine prior to the cycle-specific reload design if the previously used values for key safety parameters remain applicable for the transition to the 422V+ fuel upgrade and plant uprating. This will allow the majority of any safety analysis re-evaluations/re-analyses to be completed prior to the cycle specific design analysis.

No changes to the Westinghouse nuclear design philosophy, methods or models are necessary because of the transition to 422V+ fuel. The reload design philosophy includes the evaluation of the reload core key safety parameters which comprise the nuclear design dependent input to the USAR (Reference 1) safety evaluation for each reload cycle (Reference 2). These key safety parameters will be evaluated for each KNPP reload cycle. If one or more of the parameters fall outside the bounds assumed in the reference safety analysis, the affected transients will be re-evaluated/re-analyzed using standard methods and the results documented in the reload safety evaluation (RSE) for that cycle.

The 0.422-inch OD fuel rod has had extensive nuclear design and operating experience with the 14x14 STD fuel assembly design, which has extensive history in Point Beach Units 1 and 2 and KNPP (prior to Framatome/ANP fuel). ZIRLO material has also had extensive nuclear design and operating experience with other fuel assembly designs. These changes have a negligible effect on the use of standard nuclear design analytical models and methods to accurately describe the neutronic behavior of the 422V+ fuel (Reference 17).

3.4 DESIGN EVALUATION - PHYSICS CHARACTERISTICS AND KEY SAFETY PARAMETERS

Multiple cycles of core models were established to model the transition to a full 422V+ fueled core. These models incorporate: 422V+ style LPD ZIRLO mid-grids; ZIRLO clad fuel rods; ZIRLO fabricated guide thimble tubes and instrumentation tubes; assembly dimensional and fuel rod design modifications; and plant uprating.

Typical loading patterns were developed based on projected energy requirements or approximately 500 effective full-power days (EFPDs) for KNPP. These models are not intended to represent limiting loading

patterns. They were developed with the intent to show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing actual reload cores. Four core models were developed and used for the majority of calculations performed here.

The first "transition" cycle model is used to capture the initial and predominant transition effects. A second "transition" core model and a third "all 422V+" core model were developed to capture the core characteristics when a full core of 422V+ fuel is present at uprated conditions.

The fuel loading and assembly exposures at beginning of cycle (BOC) and end of cycle (EOC) are summarized for the first "transition" and third "all 422V+" models. Assembly power distributions at BOC, middle of cycle (MOC), and EOC are also provided for each model. These are contained in Figures 3-1 and 3-2 for the "transition" model and in Figures 3-3 and 3-4 for the "all 422V+" model. Comparisons of key core parameters versus cycle length for the models are provided in Figures 3-5 through 3-8.

Table 3-1 provides the key safety parameters' ranges. The changes in fuel design and plant uprating were accounted for in the reload transition core analysis.

3.5 DESIGN EVALUATION - POWER DISTRIBUTIONS AND PEAKING FACTORS

The current radial peaking factor limit allows the concept of low leakage fuel management to be extended by placing additional burned fuel on the periphery of the core. The reduction in power in the peripheral assemblies is offset by increased power in the remaining assemblies. This increased radial peaking is accommodated by the current radial and total peaking factor limits.

Figure 3-7 shows a comparison of the radial peaking factors between the core models used. The $F_Q^N(Z)$ (total peaking factor) limit will be 2.50 for the 422V+ fuel. A comparison of the $F_Q^N(Z)$ without uncertainty versus cycle length for each of the core models used is provided in Figure 3-8.

Beyond the power distribution impacts already mentioned, other changes to the core power distributions and peaking factors are the result of the normal cycle-to-cycle variations in core loading patterns. The normal methods of feed enrichment variation and fresh burnable absorbers will be employed to control peaking factors. Compliance with the peaking factor Technical Specifications can be assured using these methods.

3.6 TECHNICAL SPECIFICATION CHANGES RELATIVE TO NUCLEAR DESIGN

The Technical Specification changes which impact the nuclear design for 422V+ fuel are modifications to the protection trip setpoints (summarized in Section 5.1) and include non-uprated and uprated power conditions.

The following $F^N_{\Delta H}$ and $F_Q^N(Z)$ values have been considered in the reload transition analysis to appropriately bound the transition and full 422V+ cores:

$$F^N_{\Delta H} = 1.70 \times [1 + 0.3(1-P)],$$

$$F_Q^N = 2.50/P \times K(Z) \quad P > 0.5$$

where P is the fraction of full power and K(Z) is defined in the COLR. These peaking factor values allow for ease of transition to the 422V+ core and a greater degree of fuel management flexibility in reducing feed assemblies.

3.7 NUCLEAR DESIGN EVALUATION CONCLUSIONS

The key safety parameters were evaluated for KNPP as it transitions to an all 422V+ core. The values of the key safety parameters were also considered for the core uprated conditions.

Power distributions and peaking factors will show normal variations experienced with different loading patterns. The usual methods of enrichment and burnable absorber usage will be employed in the transition and full 422V+ cores to ensure compliance with the peaking factor Technical Specifications.

Table 3-1 Range of Key Safety Parameters

Safety Parameter	Analysis Values
Reactor Core Power (MWt)	1772
Vessel Average Coolant Temp. Hot Full Power (HFP) (°F)	558.1 to 574.0[1]
Coolant System Pressure (psia)	2250
Most Positive MTC (pcm/°F)	+5
Most Positive MDC ($\Delta K/g/cm^3$)	0.50
Doppler Temperature Coefficient (pcm/°F)	-2.90 to -0.91
Doppler Only Power Coefficient (pcm/%Power)	
Least Negative, Hot Zero Power (HZP) to HFP	-12.0 to -7.50
Most Negative, HZP to HFP	-24.0 to -14.0
Beta-Effective	0.0043 to 0.0072
Normal Operation $F^N_{\Delta H}$ (with uncertainties)	1.70
Shutdown Margin (%$\Delta\rho$)[2]	2.00
Normal Operation $F_Q^N(Z)$[3]	2.50

Notes:
1. Constant temperature program assumed during nominal depletion.
2. Based on fuel related input assumptions .
3. See Technical Specification , Figure 3.2-2.

	1	2	3	4	5	6	7
1	Twice B (1,2) 0' 37063 52778	Once B (4,5) +180' 18591 39557	Once B (2,6) +90' 18072 40663	Once B (6,2) +90' 18073 40797	Once B (1,6) 0' 19918 42017	Once B (5,4) 0' 18591 36629	Twice B (6,4) +180' 29068 36717
2	Once B (4,5) +270' 18591 39557	Once B (1,4) 0' 21365 41732	Feed A Feed 0 24331	Once B (3,5) +180' 20503 41917	Feed A Feed 0 24809	Feed B Feed 0 19662	Twice B (2,5) +180' 38069 44379
3	Once B (2,6) +180' 18072 40663	Feed A Feed 0 24275	Twice B (4,6) +180' 29014 47844	Once B (2,3) 0' 21261 42263	Feed B Feed 0 24004	Twice B (3,6) +180' 31302 43229	
4	Once B (6,2) +180' 18073 40797	Once B (5,3) +180' 20501 41872	Once B (3,2) 0' 21206 42214	Feed A Feed 0 24694	Feed B Feed 0 21201	Twice B (4,3) +270' 36296 43921	
5	Once B (6,1) 0' 19918 42017	Feed A Feed 0 24769	Feed B Feed 0 23981	Feed B Feed 0 21165	Twice B (7,1) +180' 29854 39487		
6	Once B (5,4) +90' 18591 36629	Feed B Feed 0 19612	Twice B (6,3) +180' 31335 43239	Twice B (3,4) +90' 36319 43919			
7	Twice B (6,4) +270' 29068 36717	Twice B (5,2) +180' 38034 44351					

Region
(Shuffle) Rotation
BOC Burnup
EOC Burnup

Figure 3-1 Transition Cycle Loading Pattern with BOC and EOC Assembly Burnups

	1	2	3	4	5	6	7
1	Twice B 0.928 0.790 0.852	Once B 1.237 1.081 1.105	Once B 1.285 1.185 1.166	Once B 1.315 1.187 1.164	Once B 1.268 1.159 1.139	Once B 0.990 0.934 0.983	Twice B 0.377 0.382 0.470
2	Once B 1.237 1.081 1.105	Once B 1.143 1.063 1.076	Feed A 1.263 1.399 1.364	Once B 1.199 1.128 1.107	Feed A 1.356 1.422 1.356	Feed B 1.064 1.095 1.136	Twice B 0.309 0.318 0.386
3	Once B 1.285 1.185 1.166	Feed A 1.259 1.395 1.363	Twice B 1.014 0.997 0.995	Once B 1.150 1.113 1.089	Feed B 1.256 1.383 1.330	Twice B 0.597 0.625 0.671	
4	Once B 1.315 1.187 1.164	Once B 1.195 1.126 1.106	Once B 1.150 1.114 1.090	Feed A 1.349 1.423 1.336	Feed B 1.206 1.200 1.162	Twice B 0.382 0.397 0.439	
5	Once B 1.268 1.159 1.139	Feed A 1.352 1.419 1.355	Feed B 1.253 1.381 1.329	Feed B 1.203 1.197 1.161	Twice B 0.501 0.501 0.532		
6	Once B 0.990 0.934 0.983	Feed B 1.060 1.092 1.135	Twice B 0.595 0.623 0.670	Twice B 0.380 0.396 0.438			
7	Twice B 0.377 0.382 0.470	Twice B 0.309 0.318 0.387					

Region
BOC Power
MOC Power
EOC Power

Figure 3-2 Transition Cycle BOC, MOC, and EOC Assembly Power Distributions

	1	2	3	4	5	6	7
1	Once B (1,1) 0' 23863 42006	Once B (5,3) +90' 23409 42711	Once B (3,5) +180' 23276 44151	Once B (6,2) +270' 19218 41401	Once B (4,5) 0' 20134 42346	Once B (2,6) 0' 19122 37677	Once B (5,4) +90' 20223 30064
2	Once B (5,3) +180' 23409 42711	Twice B (7,1) +90' 32937 50886	Feed A Feed 0 23688	Once B (2,5) 0' 24533 45389	Feed A Feed 0 24718	Feed B Feed 0 19702	Twice B (4,1) 0' 41847 48484
3	Once B (3,5) +270' 23276 44151	Feed A Feed 0 23736	Once B (4,4) 0' 24300 44685	Once B (3,2) 0' 24345 44911	Feed A Feed 0 22795	Twice B (1,2) 0' 42380 52899	
4	Once B (6,2) 0' 19218 41401	Once B (5,2) 0' 24545 45406	Once B (2,3) 0' 24424 44968	Feed A Feed 0 23971	Feed A Feed 0 19668	Twice B (3,1) 0' 45348 51856	
5	Once B (4,5) +90' 20134 42346	Feed A Feed 0 24657	Feed A Feed 0 22769	Feed A Feed 0 19651	Twice B (3,4) +270' 45071 53007		
6	Once B (2,6) +90' 19122 37677	Feed B Feed 0 19698	Twice B (5,1) +180' 42990 53448	Twice B (3,3) 0' 45476 51959			
7	Once B (5,4) +180' 20223 30064	Twice B (6,1) +90' 37390 44390					

Region
(Shuffle) Rotation
BOC Burnup
EOC Burnup

Figure 3-3 All 422 V+ Loading Pattern with BOC and EOC Assembly Burnups

	1	2	3	4	5	6	7
1	Once B 1.044 0.989 1.050	Once B 1.094 1.066 1.107	Once B 1.181 1.170 1.164	Once B 1.333 1.236 1.200	Once B 1.354 1.238 1.187	Once B 1.085 1.021 1.046	Once B 0.534 0.530 0.609
2	Once B 1.094 1.066 1.107	Twice B 0.980 1.008 1.036	Feed A 1.174 1.375 1.355	Once B 1.217 1.167 1.140	Feed A 1.362 1.416 1.344	Feed B 1.078 1.102 1.129	Twice B 0.350 0.357 0.416
3	Once B 1.181 1.170 1.164	Feed A 1.177 1.378 1.356	Once B 1.146 1.147 1.136	Once B 1.200 1.147 1.121	Feed A 1.235 1.303 1.264	Twice B 0.570 0.581 0.625	
4	Once B 1.333 1.236 1.200	Once B 1.217 1.168 1.141	Once B 1.198 1.146 1.120	Feed A 1.347 1.365 1.303	Feed A 1.195 1.091 1.073	Twice B 0.358 0.353 0.392	
5	Once B 1.354 1.238 1.187	Feed A 1.356 1.413 1.343	Feed A 1.232 1.301 1.263	Feed A 1.193 1.090 1.073	Twice B 0.459 0.430 0.462		
6	Once B 1.085 1.021 1.046	Feed B 1.076 1.101 1.130	Twice B 0.566 0.577 0.623	Twice B 0.356 0.352 0.391			
7	Once B 0.534 0.530 0.609	Twice B 0.370 0.377 0.439					

Region
BOC Power
MOC Power
EOC Power

Figure 3-4 All 422 V+ Cycle BOC, MOC, and EOC Assembly Power Distributions

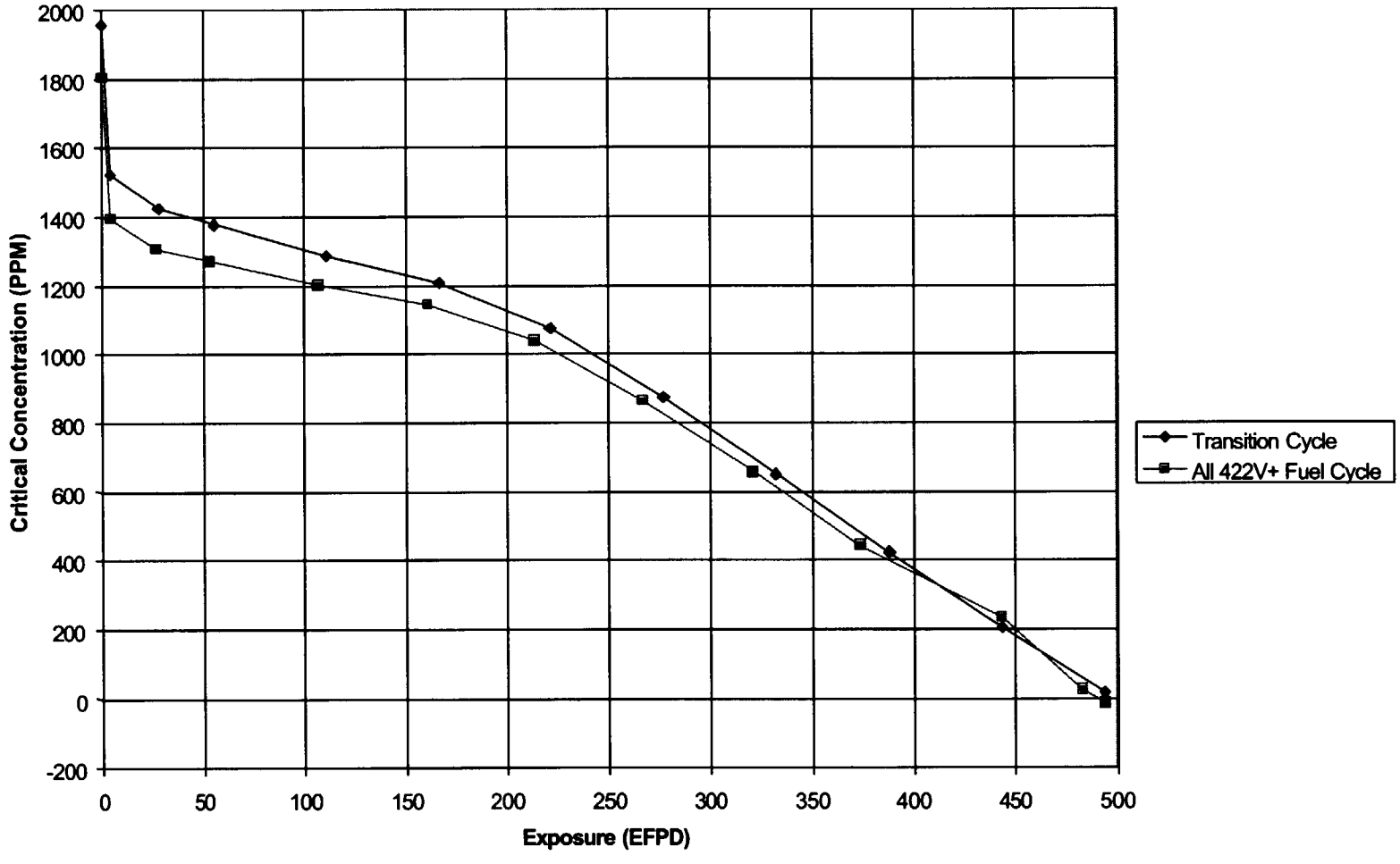

Figure 3-5 Critical Boron Concentration Comparison versus Cycle Burnup

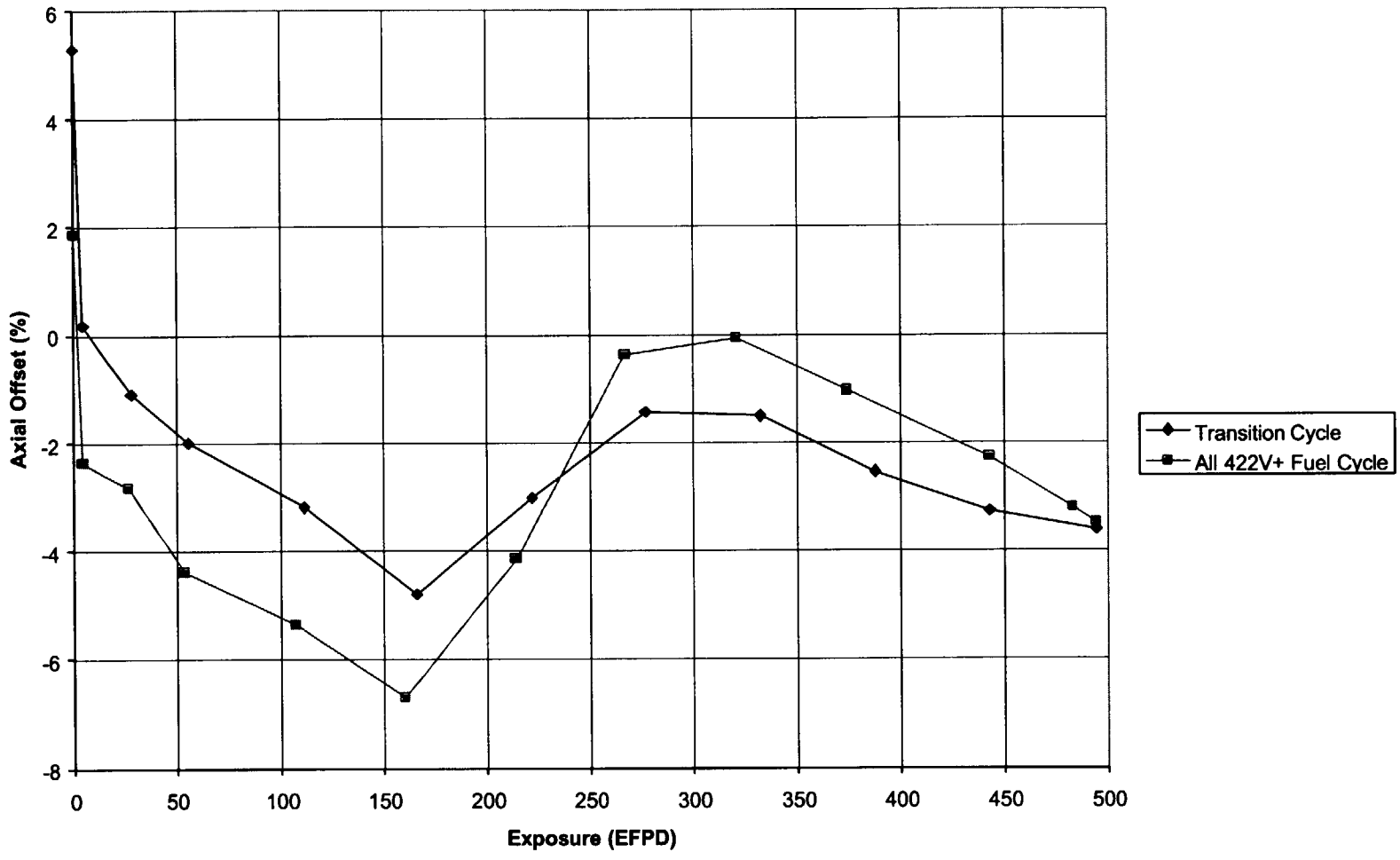

Figure 3-6 Axial Offset Comparison versus Exposure

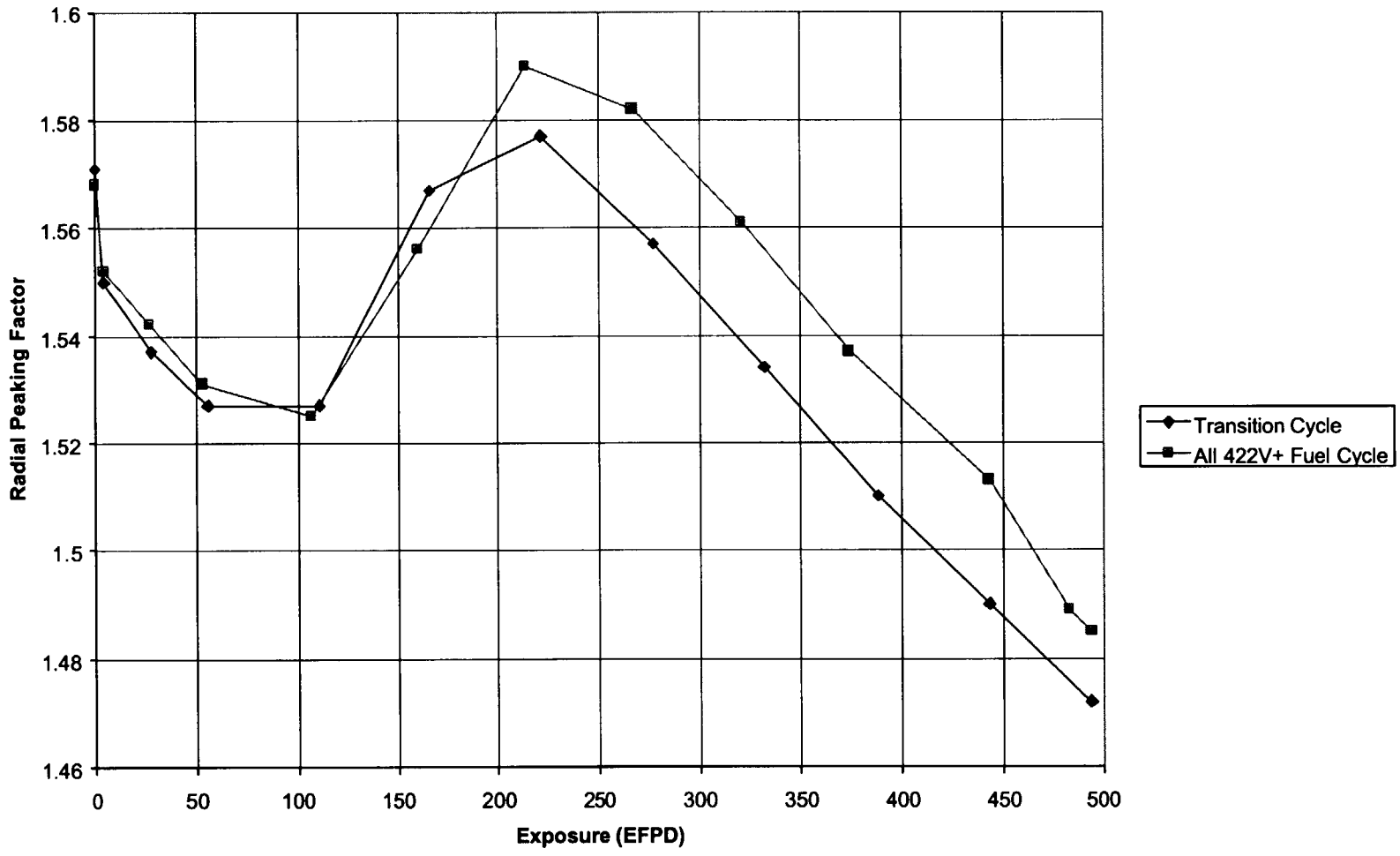

Figure 3-7 Radial Peaking Factor ($\Phi^N_{\Delta H}$) Comparison versus Exposure

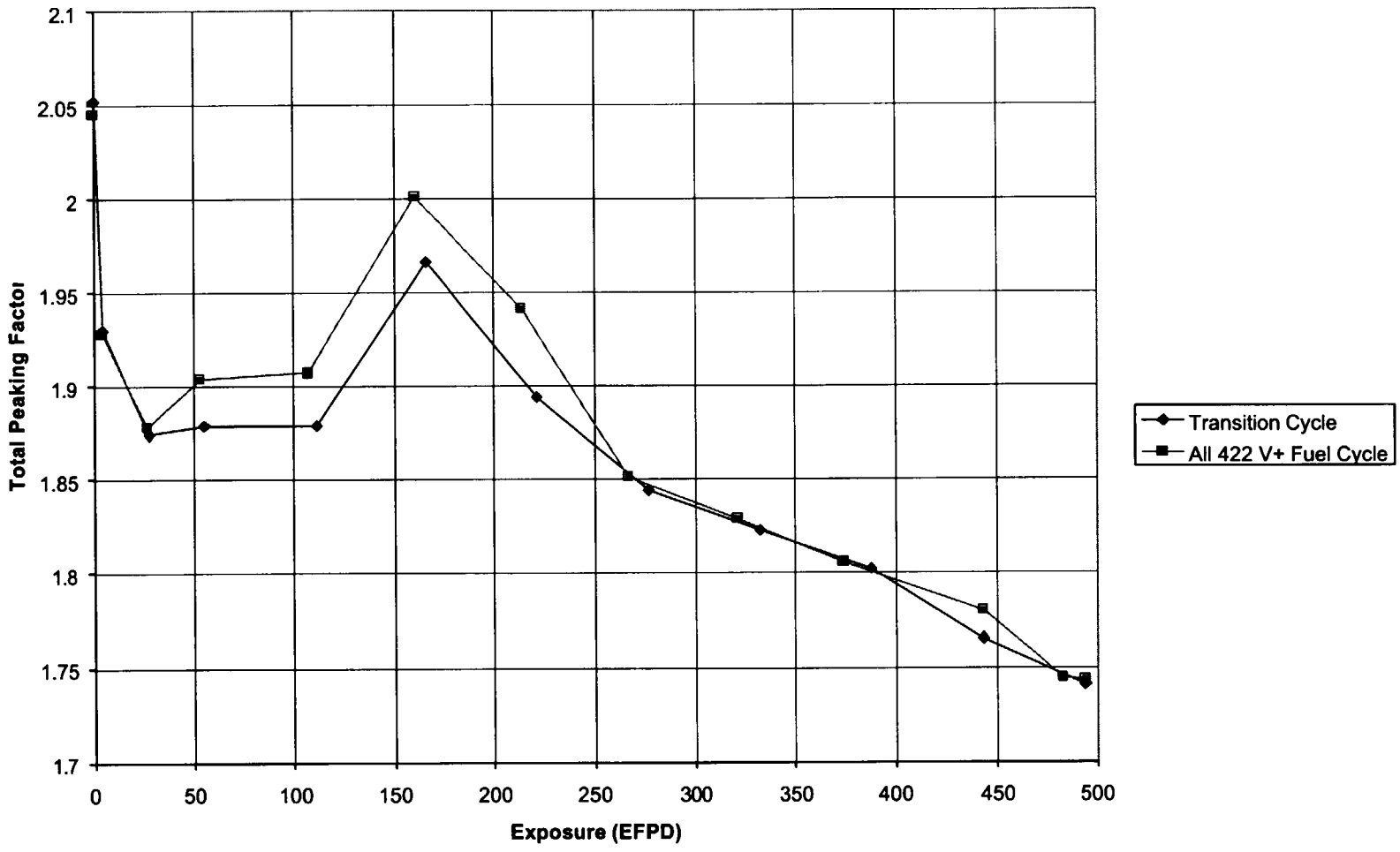

Figure 3-8 Total Peaking Factor ($F_Q^N(Z)$) Comparison versus Exposure

4 THERMAL AND HYDRAULIC DESIGN

4.1 INTRODUCTION AND SUMMARY

This section describes the calculational methods used for the thermal-hydraulic analysis, the departure from nucleate boiling (DNB) performance, and the hydraulic compatibility during the transition from an all Framatome/ANP 14x14 Heavy fuel through mixed-fuel cores to an all 422V+ core. The following hardware changes were made:

- 0.424-inch OD fuel rod to 0.422-inch OD fuel rod
- Current 14x14 Zircaloy-4 mid-grid to 14x14 ZIRLO™ mid-grid
- 0.424-inch OD instrumentation tubes to 0.422-inch OD instrumentation tubes
- 0.541-inch guide tubes to 0.526-inch guide tubes

Due to the nature of 14x14 VANTAGE+ with PERFORMANCE features, full-scale hydraulic tests (Reference 64) were performed on the 14x14 Westinghouse (422V+) fuel assembly design to confirm pressure loss compatibility with the Framatome/ANP 14x14 fuel design. Based on the overall core pressure loss, the 14x14 Westinghouse 422V+ fuel assembly design pressure drop is approximately 10 percent higher than the Framatome/ANP 14x14 design. Also, a comparison of the 14x14 Westinghouse 422V+ fuel assembly design to the Framatome/ANP 14x14 design in terms of fuel assembly crossflow velocities due to grid pressure drop mismatch, was made. In addition, a baseline for this comparison was established for a Westinghouse transition core, since Westinghouse has successfully transitioned from the 14x14 STANDARD fuel design to the 14x14 optimized fuel assembly (OFA) design. The results of the crossflow analyses show that the transition from the Framatome/ANP fuel to the 14x14 Westinghouse 422V+ fuel assembly design for KNPP was bounded by the Westinghouse transition core operational experience base. Therefore, the design criteria are satisfied. The 422V+ design allows power uprating at the current $F^N_{\Delta H}$ limit of 1.70 (discussed further in Section 4.2). Table 4-1 shows a comparison of the previous thermal-hydraulic design parameters and the new thermal-hydraulic design parameters that were used in this analysis. The thermal-hydraulic design criteria and methods remain the same as those approved for Point Beach (Reference 66) as described in the following sections. All thermal-hydraulic design criteria are satisfied for the KNPP Fuel Upgrade/Plant Uprating (FU/PU) Program.

4.2 METHODOLOGY

The thermal-hydraulic analysis of the 422V+ fuel used in KNPP is based on the Revised Thermal Design Procedure (RTDP) (Reference 15) and the WRB-1 DNB correlation (Reference 32). The departure from nucleate boiling (DNB) analysis of the core containing 422V+ fuel assemblies has been shown to be valid with the WRB-1 DNB correlation (References 32 and 66), RTDP (Reference 15), and the VIPRE W Modeling (Reference 33). The W-3 correlation and Standard Thermal Design Procedure (STDP) are still used when any one of the conditions are outside the range of the WRB-1 correlation (that is, pressure, local mass velocity, local quality, heated length, grid spacing, equivalent hydraulic diameter, equivalent heated hydraulic diameter, and distance from last grid to critical heat flux (CHF) site) and RTDP (that is, the statistical variance is exceeded on power, T_{IN}, pressure, flow, bypass, $F^N_{\Delta H}$, $F^E_{\Delta H,1}$, and F^E_Q).

The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. The approval, by the Nuclear Regulatory Commission (NRC), that a 95/95 correlation limit DNBR of 1.17 is appropriate for the 14x14 OFA fuel assemblies has been documented (Reference 36). Furthermore, it has been shown that the use of the WRB-1 correlation with a 95/95 correlation limit departure from nucleate boiling rate (DNBR) of 1.17 is appropriate for the 14x14 422V+ fuel assemblies. The WRB-1 correlation has been accepted (Reference 66) by the NRC for the Point Beach Units with 14x14 422V+ fuel.

With RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor. This factor is used to define the design limit DNBR that satisfies the DNB design criterion (that is, a plant specific design limit is that value that accounts for the RTDP uncertainties above the correlation DNBR limit). The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent (at 95-percent confidence level) for any Condition I or II event (that is, normal operation or anticipated operational occurrences). Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values. For cases where conditions fall outside the bounds of the RTDP methodology (that is, the statistical variance is exceeded on power, T_{IN}, pressure, flow, bypass, $F^N_{\Delta H}$, $F^E_{\Delta H,1}$, and F^E_Q), STDPs are used and the associated analyses are performed using input parameters with their uncertainties included.

The uncertainties included in the combined peaking factor uncertainty are:

- The nuclear enthalpy rise hot channel factor, ($F^N_{\Delta H}$)
- The enthalpy rise engineering hot channel factor, ($F^E_{\Delta H}$)
- Uncertainties in the VIPRE-W and transient codes
- Uncertainties based on surveillance data associated with vessel coolant flow, core power, coolant temperature, system pressure, and effective core flow fraction (that is, bypass flow)

The increase in DNB margin is realized when nominal values of the peaking and hot channel factors are used in the DNB safety analyses. Table 4-2 provides a listing and description of the peaking factor uncertainties.

Instrumentation uncertainties are documented in the KNPP RTDP Instrument Uncertainty Methodology Report (Reference 16). The uncertainties have changed from those previously used for the KNPP analysis to those listed below (such as, power uprating, and the current regulatory environment have all been considered in assessing the need to increase the various plant parameter uncertainties). Both the calculated uncertainties and the uncertainties used in the safety analysis, which were used for the FU/PU analyses, are listed in Table 4-3. The instrumentation uncertainties were used in determining the DNBR design limits. It should be noted that the uncertainties used in safety analyses are slightly larger than those calculated during the RTDP uncertainty analysis. The rationale of using slightly larger values for the uncertainties ensures conservatism in determining the DNBR design limit and conservatism in the overall analysis.

For the FU/PU analysis, the design limit DNBR values for the 422V+ fuel are 1.24/1.24 for typical/thimble cells. For use in the DNB safety analyses, the design limit DNBR is conservatively increased to provide DNB margin to offset the effect of rod bow, transition core, and any other DNB penalties that may occur, and to provide flexibility in design and operation of the plant. This increase in the design limit to account for various penalties and operational issues is the generic margin that exists between the design limit and the safety analysis limit. After accounting for the plant-specific margin, the safety analysis limit for the 422V+ fuel is 1.34/1.34 (typical/thimble). These safety analysis limits are employed in the DNB analyses.

With the safety analysis limit set, the core limit lines, axial offset limit lines, and dropped rod limit lines are generated. In generating the various limit lines, the maximum $F^N_{\Delta H}$ is determined that yields acceptable results based upon the safety analysis limits. Based on generating these limit lines, the maximum $F^N_{\Delta H}$ limit that can be supported is 1.70 (including uncertainties) for the 422V+ fuel. Included uncertainty accounts for the measurement uncertainty of 4 percent (Reference 34).

$$F^N_{\Delta H} = 1.70 \times [1 + 0.3(1-P)]$$

where

P = the fraction of full power

Table 4-4 summarizes the available DNBR margin for KNPP as of the completion of this analysis. It should be noted that the DNBR margin summaries are cycle dependent and may vary from that documented here in future reload designs.

4.3 HYDRAULIC COMPATIBILITY

The 14x14 422V+ and 14x14 Framatome/ANP fuel assembly designs have been shown to be hydraulically compatible (Reference 64) based on a consistent comparison of the component loss coefficients, thus minimizing effects of fuel assembly crossflow. Refer to Section 2.2 for more discussion of crossflow. The axial grid locations, grid heights, and fuel assembly pitch and envelope for the 422V+ assembly are comparable with the Framatome/ANP design, again minimizing assembly-to-assembly crossflow. By maintaining grid-to-grid overlap between the Framatome/ANP design and the 422V+ design, excessive crossflow between assemblies is prevented. The minimal difference in loss coefficients between the two designs and the respective grid locations of the two designs have been analyzed to demonstrate that no crossflow induced vibration will result in a condition in which fretting or whirling would be induced. The fuel assembly crossflow that exists for the transition core is well within the bounding Westinghouse experience basis of transition core analysis (that is, transition cores with intermediate flow mixing (IFM) vane grids will experience the maximum crossflow situation).

A second area of hydraulic compatibility associated with higher resistance fuel assemblies (the 422V+ design) is the associated impact on lift forces. When a fuel assembly with a different hydraulic resistance is loaded into a core, it changes the flow distribution in the surrounding assemblies. In particular, if this fuel assembly has a higher value of fuel assembly loss coefficient, the surrounding assemblies (that is, lower resistance fuel assemblies - the Framatome/ANP assemblies) would see a higher average flow through them than they would in a full core situation. Thus, the lift force on these surrounding assemblies

can be expected to increase. The larger the number of high resistance fuel assemblies loaded in the core, the greater the lift force is on the lower resistance assemblies.

The 14x14 Westinghouse 422V+ fuel assembly design overall pressure loss is approximately 10 percent larger than that for the Framatome/ANP 14x14 fuel (Reference 64). The 10-percent increase in pressure loss will equate to a 10-percent increase in the fuel assembly lift force. When a majority of the core is loaded with Westinghouse fuel, the Framatome/ANP assemblies can experience, in the limiting situation, a fuel assembly pressure loss equal to the Westinghouse fuel design, an increase of 10 percent in lift force.

4.4 EFFECTS OF FUEL ROD BOW ON DNBR

The concern with regard to fuel rod bow is the potential effect on bundle power distribution and on the margin of fuel rods to DNB. Thus, the phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Fuel rod bow is the phenomenon of fuel rods bowing between mid-grids. The effect of the rod bow is to impact the channel spacing between adjacent fuel rods. With a reduced channel spacing, the potential for DNB occurring increases. Westinghouse conducted test to determine the effect of rod bow on DNB. These tests and subsequent analyses were documented in Reference 35. Currently, the maximum rod bow penalty for the OFA fuel assembly is 2.6 percent DNBR at an assembly average burnup of 24,000 MWD/MTU (References 35 and 36). For burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown due to the decrease in fissionable isotopes and the buildup of fission product inventory (Reference 37). Therefore, no additional rod bow penalty is required at burnups greater than 24,000 MWD/MTU. Based on the testing and analyses of various fuel array designs (Reference 35), including the 14x14 STANDARD, evaluations have shown that the 14x14 OFA and the 14x14 422V+ fuel assemblies will have the same rod bow penalty applied to the analysis basis as that used for 14x14 STANDARD fuel assemblies.

For the 422V+ application, the rod bow penalty will be offset with DNB margin retained between the safety analysis and design DNBR limits (refer to Table 4-3).

4.5 FUEL TEMPERATURE/PRESSURE ANALYSIS

Fuel temperatures and associated rod internal pressures have been generated (Reference 44) for the 422V+ fuel. The characteristics of the Gd fuel are such that the Gd rods would exhibit higher fuel temperatures due to an inherent lower thermal conductivity of the Gd bearing fuel pellet. In addition, increasing gadilinia enrichment results in a corresponding decrease in the fuel melting temperature. The performance criteria employed by Westinghouse for Gd rods is to assure that these rods are less limiting than the non-Gd rods, throughout life, in terms of fuel temperatures, rod internal pressures and core stored energy. This is achieved by holding down the U^{235} enrichment in the Gd rods such that the Gd rods are at sufficiently lower power throughout life. Therefore, the fuel performance parameters for the 422V+ fuel bound those for the 422V+ Gd fuel. The higher fuel rod average and surface temperatures are conservative for the accident analyses performed by Transient Analysis and LOCA groups. Refer to Figures 4-1 through 4-4. In addition, the minimum fuel average and surface temperatures are required by transient analysis. Therefore, 422V+ non-Gd fuel minimum temperatures are generated that, with the maximum fuel temperatures, form a consistent basis for transient analysis.

Fuel centerline temperatures were also generated for the 422V+ fuel. These have been provided to Core Design for future verification during reload design validation that fuel melt will not occur. The maximum Kw/ft limit for fuel melt is 22.54Kw/ft for 422V+ fuel.

In addition to the fuel temperatures and pressures, the revised core stored energy for the 422V+ fuel has been determined for use in containment analysis (refer to Section 5.3). Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperature. The local core stored energy is normalized to the local linear power level. The units for the core stored energy are full-power seconds (FPS). The new value of the core stored energy for the 422V+ fuel is 4.68 FPS.

4.6 TRANSITION CORE EFFECT

Redistribution of flow in pressurized water reactor (PWR) cores is a well documented and modeled phenomenon which occurs generally because of thermal-hydraulic fluid condition gradients within the core. In a mixed core with assemblies having different hydraulic resistances, the local hydraulic resistance differences are also a mechanism for flow redistribution. This redistribution results in the fluid velocity vector having a lateral component as well as the dominant axial component. The lateral component is commonly referred to as crossflow. The crossflow induced by local hydraulic resistance differences will typically impact the mechanical design of the fuel assemblies, as well as the safety analyses of the core. Refer to Section 2.2 for additional discussion of crossflow.

The mechanical design of the fuel assemblies in the core could be affected in two ways: 1) excitation of peripheral rods in the fuel assemblies such that wear mechanisms of fretting or whirling could exist; and 2) introduction of higher resistance assemblies will influence the lift forces on the remaining assemblies. The hydraulic compatibility of the 422V+ and Framatome/ANP fuel assemblies has been addressed in Sections 2.2 and 4.3 and found to be acceptable.

In the safety analysis, crossflow affects both LOCA and DNB results. The primary consideration for the LOCA analysis is the reduction of normalized mass velocity compared to a full core of that assembly type. DNB is affected because the flow redistribution affects both mass velocity and enthalpy distributions. With current DNB correlations, WRB-1 and W-3, the flow redistribution occurs at the location where minimum DNBR is predicted. As such, the design procedure is based on the principle that once the transition core DNBR penalty is determined, all further plant-specific analysis may proceed as if it were a full core analysis.

Transition cores are analyzed as if they were full cores of one assembly type (full 422V+), then applying the applicable transition core penalty. Penalties are a function of the number of each type of fuel assembly in the core (Reference 38), which has been approved by the NRC (Reference 39). This methodology is used to calculate the Framatome/ANP to 422V+ transition core penalties. There is no DNBR transition core penalty for the Framatome/ANP fuel. However, a penalty of less than 3.0-percent applies to the 1st cycle 422V+ fuel for the 1st transition cycle operation. The penalty is applied as follows:

> The 422V+ penalty starts at 4.22 percent (with only one 422V+ assembly in the core) and decreases to a minimum value per the equation below as the number of 422V+ fuel assemblies increases. The minimum 422V+ penalty is zero for an all 422V+ core. (In actuality, the penalty applied to the 422V+ fuel would be on the order of 2.53 percent with one-third of the core

comprised of 422V+ fuel – the first transition cycle. The minimum penalty would occur in the second transition cycle when two-thirds of the core is 422V+ fuel. This minimum penalty would be on the order of 1.0 percent.)

$$\text{DNBR Penalty 422V+ (\%)} = -5.07x + 4.22$$

The value of x in the above equation is the fraction of fuel of that type remaining in the core.

The penalty applied to the 422V+ fuel is primarily due to the slightly higher flow resistance of the design. Refer to Figure 4-5 for the curve of transition core penalty applicable to the 422V+ fuel.

4.7 BYPASS FLOW

Two different bypass flows are used in the thermal-hydraulic design analysis–thermal design bypass flow (TDBF) and best estimate bypass flow (BEPF). These two bypass flows are used in non-statistical and statistical analyses respectively. TDBF is the conservatively high core bypass flow used in calculations where the results are adversely affected by low core flow. Specifically, TDBF is used with the vessel thermal design flow (TDF) in power capability analyses which use standard (non-statistical) methods. The TDBF is also used with the vessel best estimate flow (BEF) to calculate core and fuel assembly pressure drops. BEBF is the flow that would be expected using nominal values for dimensions and operating parameters that affect bypass flow without applying any uncertainty factors. The BEBF is used in conjunction with the vessel minimum measured flow (MMF) for power capability analyses that use the revised thermal design procedure (RTDP) (statistical methodology). It is also used to calculate fuel assembly lift forces. For the KNPP analyses, the maximum permissible TDBF is 7.0 and the maximum permissible BEBF is 5.5 percent.

4.8 THERMAL-HYDRAULIC DESIGN PARAMETERS

Table 4-1 lists numerous thermal-hydraulic parameters for the current design basis at 1650 MWt core power with Framatome/ANP fuel as well as the proposed design basis at 1772 MWt core power with 422V+ fuel. Some of the parameters in Table 4-1 are used in the analysis basis as VIPRE-W input parameters while others are simply provided since they are listed in the USAR. This section identifies those parameters that are used as input parameters to the VIPRE-W model and also identifies the limiting direction of each parameter. The following parameters from Table 4-1 are used in the VIPRE-W model:

- Reactor core heat output (MWt)
- Core pressure for RTDP analyses (psia)
- Heat generated in fuel (%)
- Average heat flux (BTU/hr-ft^2)
- Nominal vessel/core inlet temperature (°F)
- $F^N_{\Delta H}$, nuclear enthalpy rise hot channel factor
- Pressurizer/core pressure (psia)
- Thermal design flow for non-RTDP analyses (gpm)
- Minimum measured flow for RTDP analyses (gpm)

In addition, the average linear power (kW/ft) is used in the PAD analyses for the fuel temperatures and other fuel rod design criteria.

The limiting direction for these parameters is shown in Table 4-5.

4.9 CONCLUSION

The thermal-hydraulic evaluation of the fuel upgrade for KNPP has shown that Framatome/ANP and 422V+ fuel assemblies are hydraulically compatible and that the DNB margin gained through use of the RTDP methodology with the WRB-1 DNB correlation is sufficient to allow for a power uprating and the implementation of the Westinghouse 422V+ fuel in KNPP. More than sufficient DNBR margin in the safety limit DNBR exists to accommodate rod bow and transition core penalties. All current thermal-hydraulic design criteria are satisfied.

Table 4-1 Kewaunee Thermal-Hydraulic Design Parameters Comparison		
Thermal-Hydraulic Design Parameters	**Current Design Value**	**Analysis Value**
Reactor Core Heat Output, MWt	1650	1772(1)
Reactor Core Heat Output, 10^6, BTU/Hr	5630	6046(1)
Heat Generated in Fuel, %	97.4	97.4
Core Pressure, Nominal - RTDP, psia	2265	2265
Pressurizer Pressure, Nominal, psia	2250	2250
Radial Power Distribution(2)	1.70[1+0.2(1-P)], Heavy 1.55[1+0.2(1-P)], STD	1.70[1+0.3(1-P)], 422V+
	where $P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$	
HFP Nominal Coolant Conditions		
Vessel TDF Rate (including bypass)		
10^6 lbm/hr	67.5	67.87
GPM	178,000	178,000
Core Flow Rate (excluding Bypass,(3) based on TDF)		
10^6 lb_m/hr	62.8	63.12
GPM	165,540	165,540
Core Flow Area, ft^2	26.7	27.1 (full-core 422V+)
Core Inlet Mass Velocity, (Based on TDF)		
10^6 lb_m/hr-ft^2	2.35	2.33

Notes:

1. The proposed power level of 1772 MWt has been used for all thermal-hydraulic design analyses. The proposed power level is thus bounding for the thermal-hydraulic design analyses since the current power level of 1650 MWt will be maintained until the power uprating is approved by the NRC. USAR updates for the FU/PU program will reflect the current power level of 1650 MWt until the power uprating request is approved.
2. Includes 4 percent measurement uncertainty (Reference 67).
3. Based on design bypass flow of 7 percent for current design value.

Table 4-1 Kewaunee Thermal-Hydraulic Design Parameters Comparison (cont.)

Thermal-Hydraulic Design Parameters (based on TDF)	Current Design Value	Analysis Value
Nominal Vessel/Core Inlet Temperature, °F	543.8	539.2
Vessel Average Temperature, °F	575.3	573.0
Core Average Temperature, °F	579.0	577.1
Vessel Outlet Temperature, °F	606.8	606.8
Core Outlet Temperature, °F	611.0	611.3
Average Temperature Rise in Vessel, °F	63.0	67.6
Average Temperature Rise in Core, °F	67.2	72.1
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	28,851	28,565
Average Heat Flux, BTU/hr-ft^2	190,071	206,165(1)
Average Linear Power, kw/ft	6.35	6.85
Peak Linear Power for Normal Operation,(2) kW/ft	15.9	17.18(1)
Peak Linear Power for Prevention of Centerline Melt, kW/ft	N/A	22.54
Pressure Drop Across Core, psi(3)		
Full core of Framatome/ANP	20.0	
Full core of 422V+	-----	23.0

Notes:

1. The proposed power level of 1772 MWt has been used for all thermal-hydraulic design analyses. The proposed power level is thus bounding for the thermal-hydraulic design analyses since the current power level of 1650 MWt will be maintained until the power uprating is approved by the NRC. USAR updates for the FU/PU program will reflect the current power level of 1650 MWt until the power uprating request is approved.
2. Based on maximum F_Q of 2.50.
3. Based on best estimate reactor flow rate of 98,900 gpm/loop.

Table 4-2 Peaking Factor Uncertainties		
$\Phi_{\Delta H} = F^N_{\Delta H} \times F^E_{\Delta H}$		
where:	$\Phi^N_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor – The ratio of the relative power of the hot rod, which is one of the rods in the hot channel, to the average rod power. The normal operation value of this is given in the plant Technical Specifications or a Core Operating Limit Report (COLR).
	$F^E_{\Delta H}$	Engineering Enthalpy Rise Hot Channel Factor – The nominal enthalpy rise in an isolated hot channel can be calculated by dividing the nominal power into this channel by the core average inlet flow per channel. The engineering enthalpy rise hot channel factor accounts for the effects of flow conditions and fabrication tolerances. It can be written symbolically as:
$F^E_{\Delta H} = f(F^E_{\Delta H,1}, F^E_{\Delta H,2}, F^E_{\Delta H\ inlet\ maldist}, F^E_{\Delta H\ redist}, F^E_{\Delta H\ mixing})$		
where:	$F^E_{\Delta H,1}$:	accounts for rod-to-rod variations in fuel enrichment and weight
	$F^E_{\Delta H,2}$:	accounts for variations in fuel rod outer diameter, rod pitch, and bowing
	$F^E_{\Delta H\ inlet\ maldist}$:	accounts for the nonuniform flow distribution at the core inlet
	$F^E_{\Delta H\ redist}$:	accounts for flow redistribution between adjacent channels due to the different thermal-hydraulic conditions between channels
	$F^E_{\Delta H\ mixing}$:	accounts for thermal diffusion energy exchange between adjacent channels caused by both natural turbulence and forced turbulence due to the mixing vane grids
The value of these factors and the way in which they are combined depends upon the design methodology used, that is, STDP or RTDP. Note that no actual combined effect value is calculated for $F^E_{\Delta H}$. These factors are accounted for by using the VIPRE-W code.		

Table 4-3 RTDP Uncertainties

Parameter	Calculated Uncertainty	Uncertainty Used in Safety Analysis
Power	±1.72% -0.32% bias	±2.0% -0.32% bias (at 1757 MWt NSSS power)
Reactor Coolant System Flow	±2.86% +0.11% bias	±4.3% +0.11% bias
Pressure	±35.1 psi 15.0 psi bias	±50.0 psi 15.0 psi bias
Inlet Temperature	±4.9°F -1.1°F (bias)	±6.0°F -1.1°F (bias)

Table 4-4 DNBR Margin Summary[1]		
Fuel Type		**422V+**
DNB Correlation		WRB-1
DNBR Correlation Limit		1.17
DNBR Design Limit	(TYP)[2]	1.24
	(THM)[3]	1.24
DNBR Safety Limit	(TYP)	1.34
	(THM)	1.34
DNBR Retained Margin[4]	(TYP)	7.46%
	(THM)	7.46%
DNBR Margin prior to 7.4% Power Uprate	(TYP)	12%
	(THM)	12%
Rod Bow DNBR Penalty		-2.6%
Instrumentation Bias Penalty	(TYP)	-3.02
	(THM)	-2.79
Transition Core DNBR Penalty		-2.53%
Available DNBR Margin	(TYP)	11.31%
	(THM)	11.54%

Notes:

1. Steam line break is analyzed using the W-3 correlation with STDP. The correlation limit DNBR is 1.45 in the range of 500 to 1000 psia. Rod withdrawal from subcritical is also analyzed using the W-3 correlation (w/o spacer factor) with STDP below the bottom NMV grid. The correlation limit DNBR is 1.30 above 1000 psia and the safety limit DNBR is 1.39 which provides 6.7% margin to cover the rod bow penalty and retain generic margin for operational issues. WRB-1 with RTDP is used for rod withdrawal from subcritical above the bottom NMV grid.
2. TYP = Typical Cell
3. THM = Thimble Cell
4. DNBR margin is the margin that exists between the safety limit and the design limit DNBRs.

Table 4-5 Limiting Parameter Direction

Parameter	Limiting Direction for DNB
$F^N_{\Delta H}$, nuclear enthalpy rise hot channel factor	maximum
Heat generated in fuel (%)	maximum
Reactor core heat output (MWt)	maximum
Average heat flux (BTU/hr-ft^2)	maximum
Nominal vessel/core inlet temperature (°F)	maximum
Core pressure (psia)	minimum
Pressurizer pressure (psia)	minimum
Thermal design flow for non-RTDP analyses (gpm)	minimum
Minimum measured flow for RTDP analyses (gpm)	minimum

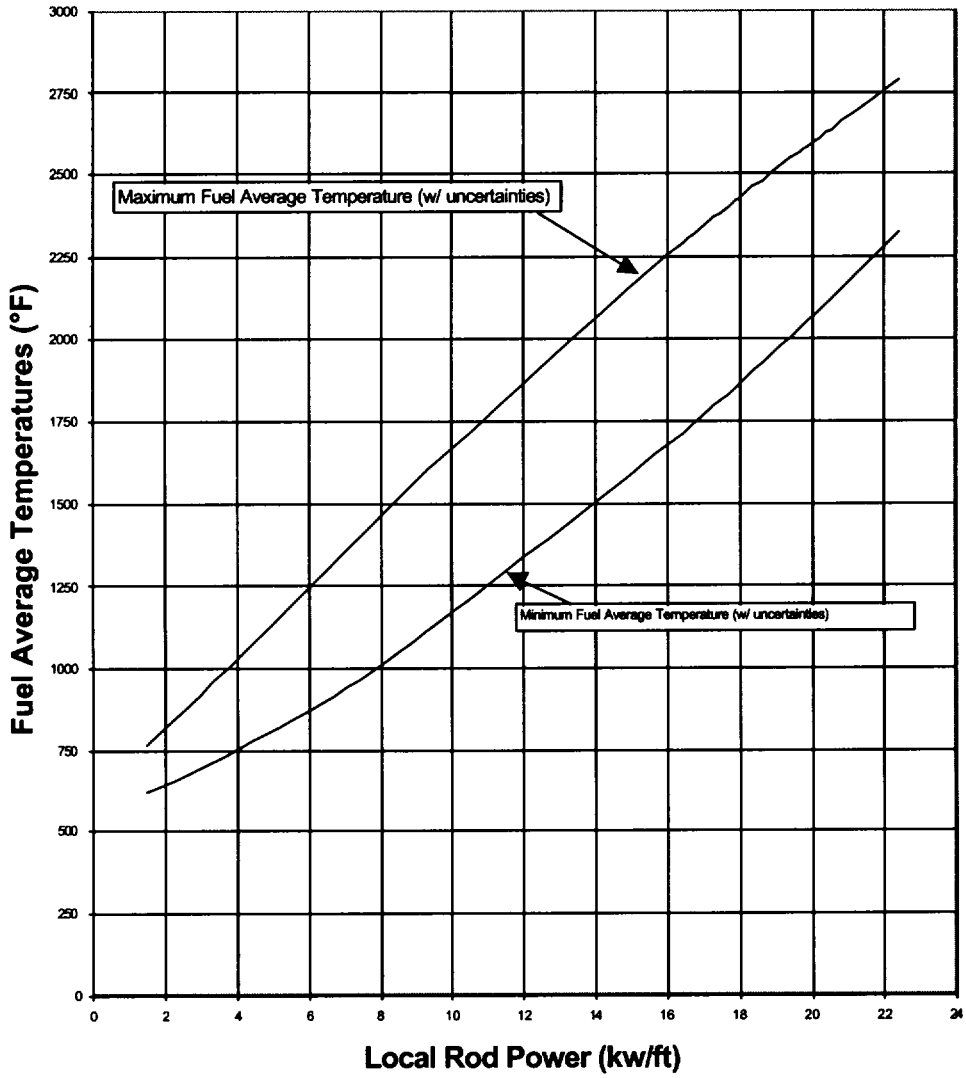

Figure 4-1 Fuel Average Temperatures

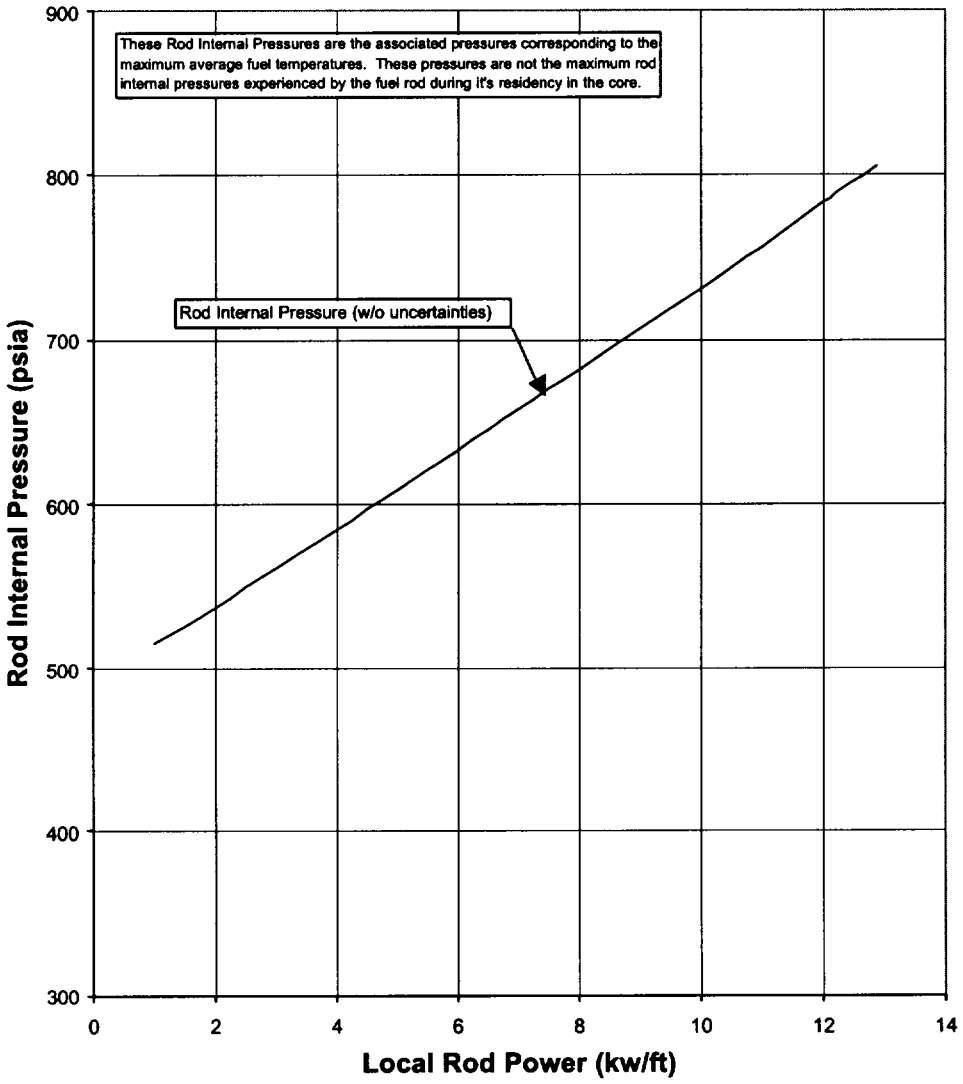

Figure 4-2 Rod Internal Pressure

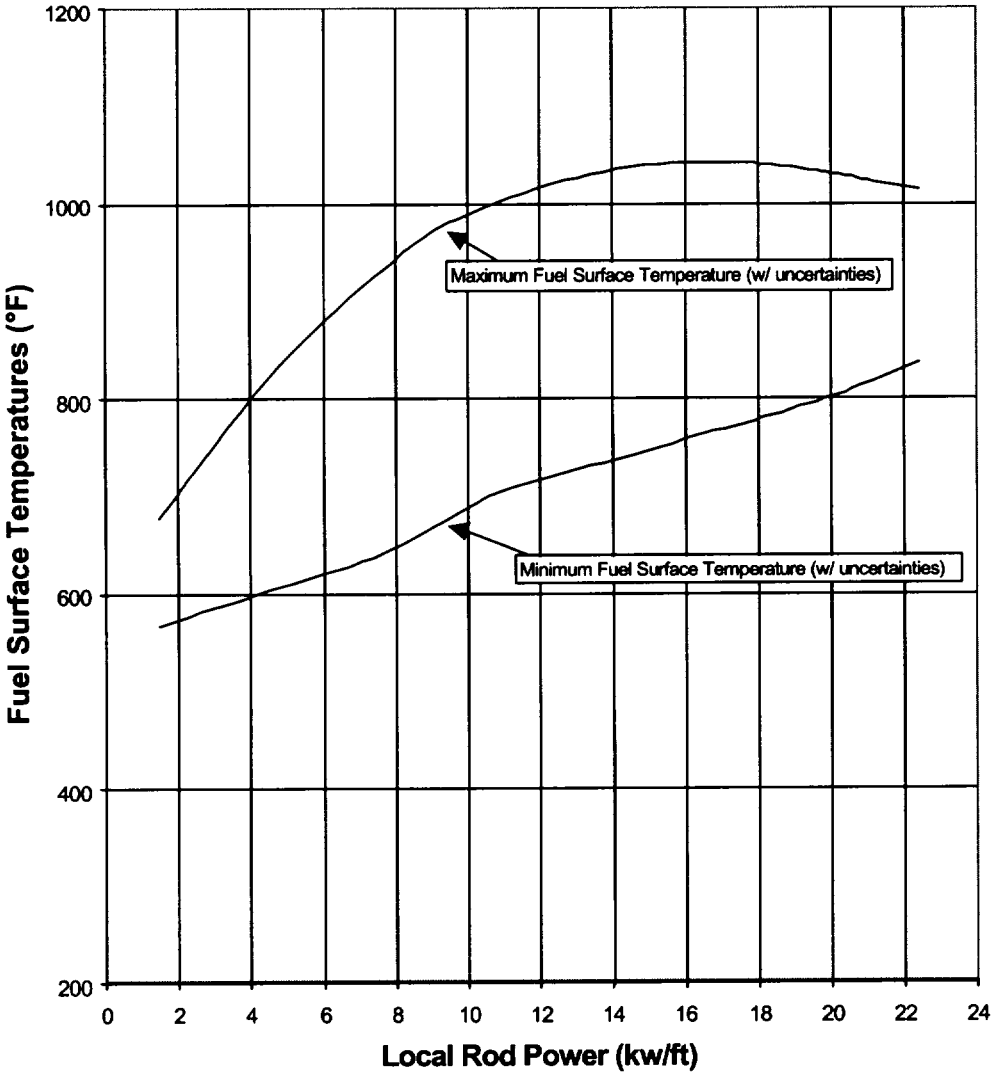

Figure 4-3 Fuel Surface Temperatures

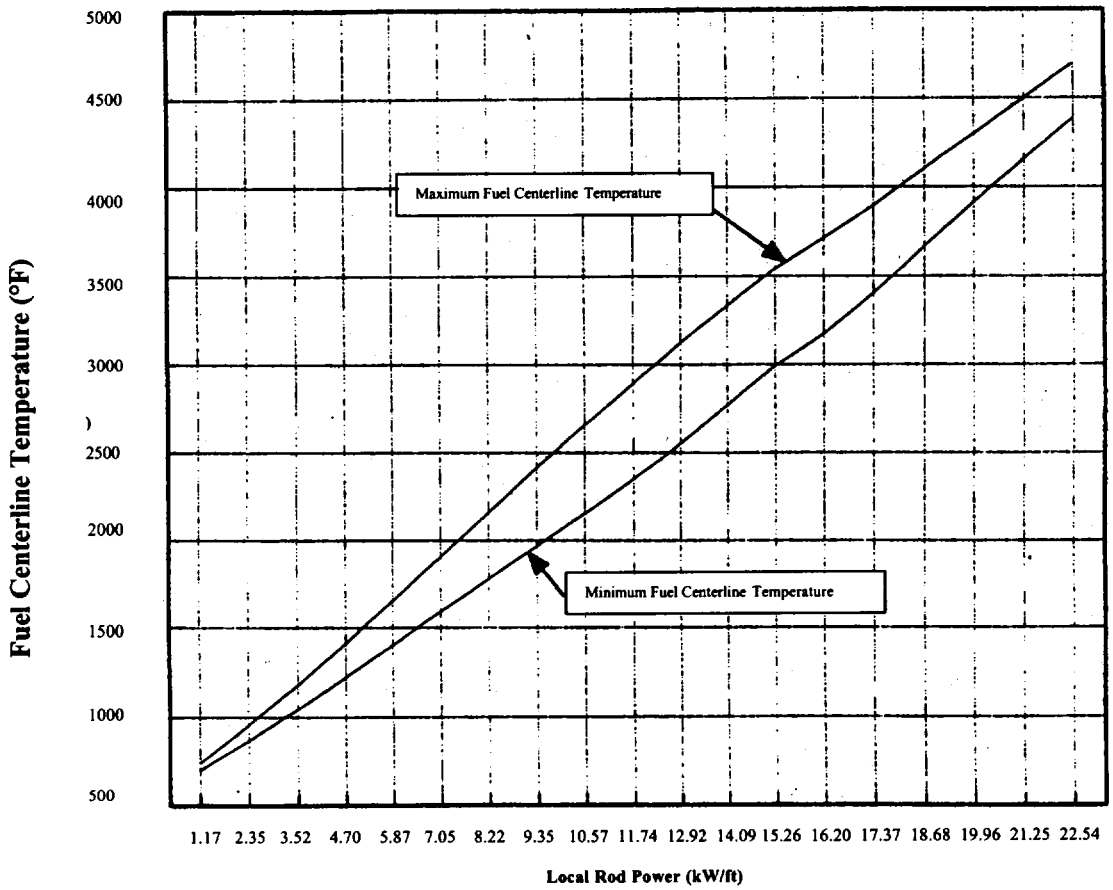

Figure 4-4 Fuel Centerline Temperatures

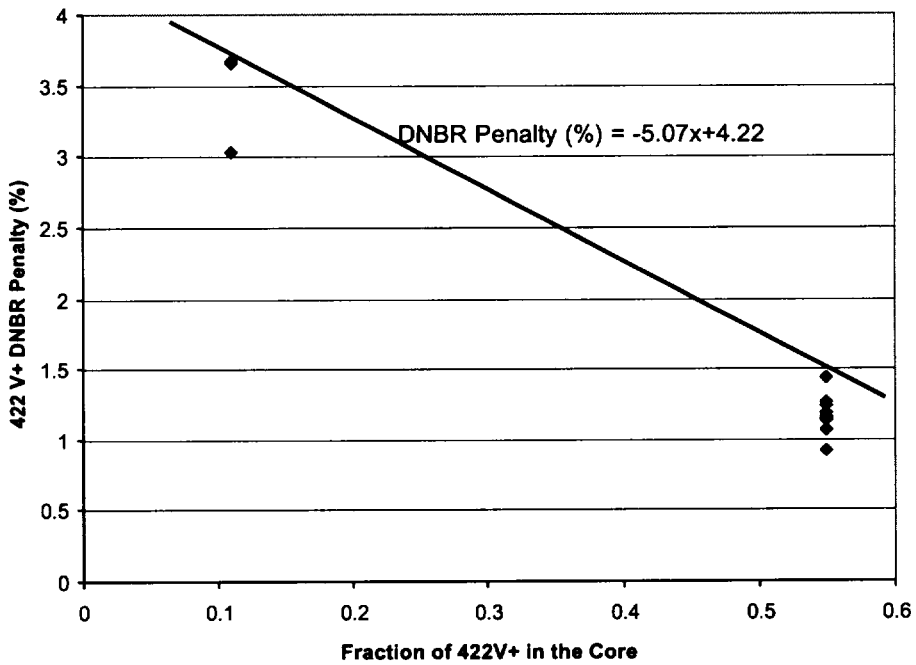

Figure 4-5 Transition Core Penalty as a Function of Fuel in the Core

5 ACCIDENT ANALYSIS

This section presents the results of analyses and evaluations of non-loss-of-coolant accident (LOCA) transients, large and small-break LOCAs, and containment integrity.

5.1 NON-LOCA TRANSIENTS

Non-LOCA transient analyses and evaluations were performed to support implementation of the 422V+ fuel transition and power upgrade programs at the Kewaunee Nuclear Power Plant (KNPP). All non-LOCA events analyzed in the KNPP Updated Safety Analysis Report (USAR) were analyzed or evaluated. Uprating program features that were considered include:

- A nuclear steam supply system (NSSS) power level of 1780 MWt (including 8 MWt of reactor coolant pump heat)
- T_{avg} range from 556.3°F to 573.0°F
- Reactor coolant system (RCS) thermal design flow (TDF) of 178,000 gpm
- Steam generator tube plugging (SGTP) up to 10 percent (uniform – no asymmetry assumed)
- Nominal operating RCS (pressurizer) pressure of 2250 psia

Computer codes used in the analyses are FACTRAN (Reference 45), RETRAN (Reference 46, LOFTRAN (Reference 47), ANC (Reference 48), TWINKLE (Reference 49), and VIPRE (Reference 50). Table 5.1-7 indicates the codes used for each analysis. The table also presents the assumed principal initial conditions and other methodology information pertinent to each event.

The results of the non-LOCA analyses and evaluations, except for the anticipated transients without scram (ATWS) event, are presented in Table 5.1-1. All acceptance criteria are met. Although the ATWS event was evaluated, it is not addressed in Table 5.1-1 because the evaluation results do not lend themselves to presentation in a tabular format. Specifically, 10 CFR 50.62(b) requires that Westinghouse pressurized water reactors (PWRs) implement the ATWS mitigating system actuation circuitry (AMSAC). AMSAC is installed at KNPP, so the requirements of 10 CFR 50.62(b) are satisfied. A diverse scram system (DSS) is installed at KNPP as a supplement to AMSAC. The NRC has approved the implementation of AMSAC and the DSS at KNPP as documented in Section 14.1.11 of the KNPP USAR. AMSAC and DSS will be maintained and operated following implementation of the 422V+ FU Program consistent with their design bases and as approved by the NRC. Consequently, required protection against postulated ATWS events will be maintained, and analysis of ATWS events is not required.

5.2 LOSS-OF-COOLANT ACCIDENTS

5.2.1 Large-Break Best-Estimate LOCA

The Westinghouse best-estimate loss-of-coolant accident analysis methodology is documented in WCAP-12945-P-A (Reference 51) and is approved by Reference 52. The methodology was extended to plants having residual heat removal (RHR) injection into the upper plenum by Reference 53, and was approved by Reference 54. Westinghouse applied this approved methodology to analyze the large-break LOCA (LBLOCA) at the uprated power level of 1772 MWt for the KNPP 422V+ Fuel Upgrade/Power Uprating (FU/PU) Program.

For the best-estimate upper plenum injection (UPI) evaluation model, References 52 and 54 requires that the plant conditions analyzed with WCOBRA/TRAC fall within the range of conditions represented by test simulations used to assess phenomena unique to UPI plants. Table 5.2.1-3 demonstrates that the KNPP analysis parameters fall within the range of the test conditions.

The transition from Framatome/ANP fuel to Westinghouse 422V+ fuel was evaluated to determine the effects of hydraulic and design differences of the two types of fuel. Evaluations of mixed cores demonstrated that the best-estimate analysis results of a full 422V+ core bound the results for transition cycles.

Table 5.2.1-2 presents the LBLOCA analysis results. All acceptance criteria are met.

5.2.2 Small-Break LOCA

Westinghouse evaluated the small-break LOCA (SBLOCA) using the NOTRUMP code (References 55 and 56), including changes described in Reference 57, and the SBLOCTA code, a small-break-specific version of the LOCTA-IV code (Reference 58).

The complete small break spectrum of 2-, 3-, and 4-inch breaks was analyzed at the uprated power level of 1772 MWt. Since the core flow during the SBLOCA is relatively slow, flow equilibrium between fuel assemblies is maintained (that is, no cross flow), and hydraulic mismatch is not a factor for the SBLOCA as was demonstrated in Reference 59. Consequently, performing a small break analysis for transition cores is not necessary, and the analysis for a full 422V+ core bounds transition cycles.

The limiting break was found to be the 3-inch break initiated at the low assumed T_{avg} value of 546.3°F. The peak clad temperature was 1030°F, local clad oxidation was less than 17 percent, and core-wide oxidation was less than 1 percent.

5.2.3 Post-LOCA Long-Term Subcriticality, Cooling Evaluation

Westinghouse's position for satisfying the requirements of 10 CFR 50.46(b)(5) is specified in WCAP-8339-NP-A (Reference 60), WCAP-8472-NP-A (Reference 61), and Technical Bulletin NSID-TB-86-08 (Reference 62). The reactor will remain shutdown on boron alone.

The post-LOCA containment mixed mean sump boron concentration was calculated for the 422V+ FU/PU Program, and a plot of containment sump boron concentration as a function of pre-trip RCS boron concentration was developed. The plot is included in the Reload Safety Analysis Checklist (RSAC) and is checked for each reload cycle to ensure that adequate boron will exist in the sump to maintain subcriticality in the long-term post-LOCA.

In addition to confirming post-LOCA subcriticality, maintenance of core cooling after swtichover to the sump recirculation mode must be demonstrated. For the LBLOCA, core cooling is assured by confirming that there is sufficient emergency core cooling system (ECCS) flow to offset core boiloff and boiling in the downcomer and lower plenum. For the SBLOCA, potential effects of ECCS flow interruptions and/or enthalpy changes at switchover to the recirculation mode are considered as part of the SBLOCA analysis.

5.2.4 Post-LOCA Boron Build-Up Analysis and Long-Term Post-LOCA Cooling

For the KNPP, the post-LOCA core boron buildup analysis was most recently performed as part of the Steam Generator Replacement (SGR) and T_{avg} Operating Window Program (Reference 70). This core boron buildup analysis was redone for the 422V+ Fuel Upgrade and Power Uprate (FU/PU) Program because of the effects of the higher uprated core power and associated decay heat, plus other changes to analysis input parameters. The boron precipitation reanalysis for the KNPP for the 422V+ FU/PU Program demonstrates the acceptability of an 18-hour criteria for achieving LHSI post-LOCA.

5.3 CONTAINMENT ANALYSIS

The current licensing basis subcompartment analyses are discussed in Kewaunee Nuclear Power Plant (KNPP) Updated Safety Analysis Report (USAR) Section 5.9.2. The current containment integrity analyses are discussed in USAR Section 14.3.4. The containment subcompartment analysis is performed to demonstrate the integrity of containment internal structures when subjected to dynamic, localized pressurization effects that could occur during the very early time period following a design basis loss-of-coolant accident (LOCA). The containment integrity analysis is performed to demonstrate that the containment, containment structures, and containment cooling systems are adequate to mitigate the consequences of a hypothetical large-break LOCA (LBLOCA). The resultant containment pressures must not exceed the KNPP containment design pressure of 46 psig. The impact of replacing the current fuel design with the 422V+ fuel design on these containment-related analyses for KNPP has been determined.

5.3.1 Short-Term LOCA Mass & Energy/Subcompartment Analysis

The initial RCS pressure and temperatures and the core-stored energy have the potential of changing as a result of the fuel change. For this analysis there is no change in the initial RCS operating pressure. The current analysis of record considered an initial pressure of 2250 psia plus uncertainty. Any change in core stored energy would have no affect on the releases because of the short duration of the postulated accident. Therefore, the only effect that needs to be addressed is the potential for decreased RCS coolant temperatures. Decreased RCS temperatures result in higher density break flow, creating a more adverse situation.

The hot leg temperature for the 422V+ Fuel Upgrade/Power Uprating (FU/PU) program is 2.8°F higher than the current design basis analysis, and the cold leg temperature is 2.0°F lower. KNPP is, however, approved for leak before break (LBB), so RCS piping and surge line ruptures have been eliminated from consideration. Only the large branch line nozzles must be considered. These include the accumulator lines, the pressurizer spray line, and the residual heat removal (RHR) line. These smaller potential breaks would be outside the reactor cavity and would produce minimal asymmetric pressurization in the cavity region. Differential pressure loadings would also be significantly reduced. For example it is estimated that peak subcompartment pressure would be reduced by at least 50 percent.

Since KNPP is licensed for LBB, the decrease in mass and energy releases associated with the smaller breaks would more than offset potential effects of a slightly decreased RCS cold leg temperature associated with the 422V+ FU/PU Program. The current licensing basis analysis remains bounding.

5.3.2 Long-Term LOCA/Containment Integrity Analysis

The current licensing basis (Reference 70) considered an initial pressure of 2250 psia plus uncertainty and an initial power of 1650 MWt plus 2-percent uncertainty, which have not changed for this evaluation. The core-stored energy for the 422V+ fuel design of 4.68 full-power seconds is less than the core-stored energy of 6.40 full-power seconds used for the SGR program for the current fuel product. The initial operating temperature window has remained essentially the same as for the current licensing basis (current licensing basis T_{avg} of 579.3°F versus T_{avg} of 579.0°F for this evaluation). The slight reduction in RCS average temperature and the reduction in core-stored energy are beneficial with respect to energy releases, and the current licensing basis from the SGR program remains bounding.

5.3.3 Conclusion

Replacing the current fuel design with the 422V+ design is acceptable with respect to containment-related analyses.

5.4 RADIOLOGICAL ANALYSIS

The radiological consequences of design basis accidents were reanalyzed in accordance with the provisions of 10 CFR 50.67, "Accident Source Term," to justify revising the Kewaunee Nuclear Power Plant (KNPP) source term for design basis radiological analyses.

Core source terms and coolant activities were calculated as part of the KNPP Steam Generator Replacement (SGR) program. Core and coolant radionuclide inventory calculations have been performed for transition from Framatome/ANP fuel to Westinghouse fuel as part the 422V+ Fuel Upgrade/Power Uprating (FU/PU) Program, and differences from the source terms calculated for the SGR program were found to be small. Consequently, the SGR source terms were increased by 10 percent to accommodate changes associated with power uprating and the 422V+ fuel design, and the resulting source terms were used for radiological analyses of the following design basis events:

- LOCA
- Locked rotor
- Rod ejection
- Main steam line break
- Steam generator tube rupture
- Fuel handling accident
- Gas decay tank rupture
- Volume control tank rupture

Reference 71 provided results of radiological analyses of the above events to the NRC, and demonstrated compliance with acceptance criteria. NRC approval of Reference 71 is expected in December 2002.

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	100 60 10 (1780 MWt – NSSS power)	186,000 (DNB) 178,000 (Pressure	573.0 (100%) 562.6 (60%) 549.6 (10%) 555.6 (8%)	2,250
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)
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 5.1-1 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	> 1.39
		Maximum fuel centerline temperature	4746°F[1]	< 4746°F
14.1.2	Uncontrolled RCCA Withdrawal at Power	Minimum DNBR (RTDP, WRB-1)	1.34	> 1.34
		Peak RCS pressure	2750 psia	< 2750 psia
		Peak MS system pressure	1208.5 psia	< 1208.5 psia
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	15 minutes	> 15 minutes
	(during startup)	Minimum time to loss of shutdown margin	15 minutes	> 15 minutes
	(during refueling)	Minimum time to loss of shutdown margin	30 minutes	> 30 minutes
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 percent 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 Non-LOCA Analysis Limits and Analysis Results (cont.)

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	> 1.34
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	> 1.34
	Locked Rotor	Rods-in-DNB (RTDP, WRB-1)	See Section 5.4	
		Peak RCS pressure	2750 psia	< 2750 psia
		Peak cladding temperature	2700°F	< 2700°F
		Maximum Zirc-water reaction	16%	< 16%
14.1.9	Loss of External Electrical Load	Minimum DNBR (RTDP, WRB-1)	1.34	> 1.34
		Peak RCS pressure	2750 psia	< 2750 psia
		Peak MS system pressure	1208.5 psia	< 1208.5 psia
14.1.10	Loss of Normal Feedwater	Maximum pressurizer water volume	1010.1 ft^3	960 ft^3
14.1.11	Loss of AC Power to the Plant Auxiliaries	Maximum pressurizer water volume	1010.1 ft^3	698 ft^3
14.2.5	Steam Line Break (Core response only)	Minimum DNBR (non-RTDP, W-3)	1.472	> 1.472

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

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	360 Btu/lbm	< 360 Btu/lbm
		Maximum fuel melt	10%	< 10%
		Maximum cladding average temperature	2700°F	< 2700°F
		Maximum Zirc-water reaction	16%	< 16%
	(BOC-HFP)	Maximum fuel pellet average enthalpy	360 Btu/lbm	< 360 Btu/lbm
		Maximum fuel melt	10%	< 10%
		Maximum cladding average temperature	2700°F	< 2700°F
		Maximum Zirc-water reaction	16%	< 16%
	(EOC-HZP)	Maximum fuel pellet average enthalpy	360 Btu/lbm	< 360 Btu/lbm
		Maximum fuel melt	10%	< 10%
		Maximum cladding average temperature	2700°F	< 2700°F
		Maximum Zirc-water reaction	16%	< 16%
	(EOC-HFP)	Maximum fuel pellet average enthalpy	360 Btu/lbm	< 360 Btu/lbm
		Maximum fuel melt	10%	< 10%
		Maximum cladding average temperature	2700°F	< 2700°F
		Maximum Zirc-water reaction	16%	< 16%

Table 5.2.1-3 KNPP Conditions Analyzed with WCOBRA/TRAC Compared to Best-Estimate UPI Test Conditions

Condition	BE UPI Test	KNPP
Core Power, MWt	1980	1772
Low Power Region Average Linear Heat Rate (kW/ft)	6.9	1.4 – 4.1
Peak Linear Heat Rate (kW/ft)	17.0	16.995*

*Note that this is a higher kw/ft than the Technical Specifications would allow.

Table 5.2.1-2 Best-Estimate UPI Large-Break LOCA Results		
	Value	**Criteria**
50th Percentile PCT (°F)	<1760	N/A
95th Percentile PCT (°F)	<2084	<2200
Maximum Cladding Oxidation (%)	8.44	<17
Maximum Hydrogen Generation (%)	0.74	<1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long-Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

6 SYSTEMS AND COMPONENTS ANALYSIS

6.1 MARGIN-TO-TRIP ANALYSIS

6.1.1 Introduction

The following reactor trip and Engineered Safety Features (ESF) actuation systems are active during at-power operation. The reactor trips, as described in Final Safety Analysis Report (USAR) Section 7.2.2 (Reference 1), are high- and low-power range nuclear flux, overtemperature ΔT (OTΔT), overpower ΔP (OPΔT), low pressurizer pressure, high pressurizer pressure, low reactor coolant flow, reactor coolant pump breakers (underfrequency, undervoltage, bus fault), safety injection signal (actuation), turbine-generator trip, steam/feedwater flow mismatch, low-low steam generator water level, intermediate range nuclear flux, and source range nuclear flux. The other reactor trips are either manually actuated or are not active during at power operation.

As described in USAR Section 7.2.2, the ESF actuation system automatically initiates the following subsystem of ESF when any of the following conditions exists: low pressurizer pressure (safety injection [SI]), high containment pressure (SI), low steam line pressure (SI), coincidence of SI signal, low-low T_{avg} and high steam flow (steam line isolation [SLI]), coincidence of SI signal and high-high steam flow (SLI), and high containment pressure (SLI).

As part of the KNPP Westinghouse 14X14 VANTAGE + fuel with PERFORMANCE + features (422V+) fuel upgrade (FU), the plant operating margins to various reactor trips and ESF actuation systems were evaluated. The analyses bound the following:

- Power level rating of 1657.1 MWt
- T_{avg} operating window between 554.1°F and 575.3°F for normal operating pressures of 2250 psia
- Model 54F steam generator type
- 422V+ fuel types

The Westinghouse analyses for the FU were largely performed for a bounding power uprating to 1780 MWt (defined as the Fuel Upgrade/Power Uprating Program, or FU/PU). This is a 7.4-percent power increase over the present licensed power level of 1657.1 MWt. Engineering efforts are presently underway to cover a 7.4-percent uprating condition with a T_{avg} operating window between 556.3°F and 573.0°F for normal operating pressures of 2250 psia (results in maximum T_{hot} and minimum T_{cold} no more severe than the values for the present power level of 1657.1 MWt). The margin to trip analyses were performed for this 7.4-percent power uprate condition in order to conservatively bound the FU at the present power level and to cover future fuel cycles that will encompass the PU conditions.

The OTΔT and OPΔT reactor trip setpoints were changed for the KNPP 422V+ FU Program. KNPP will determine the nominal high and low pressurizer pressure reactor trip setpoints. All other reactor trip setpoints remain unchanged. All ESF actuation setpoints remain unchanged.

A best-estimate analysis or evaluation of the following transients was performed for the control systems operability/margin to trip evaluation:

- 10-percent step load increase from 90-percent power
- 10-percent step load decrease from 100-percent power
- 50-percent load rejection from 100-percent power
- 5-percent/minute load increase from 15-percent power

The results of the margin-to-trip analyses indicate that there is adequate margin to the revised OTΔT/OPΔT reactor protection systems setpoints and that the control system is adequately stable for the FU Program, and are documented in KNPP's RTSR.

6.2 FLUID SYSTEMS (BORDER) ANALYSIS

6.2.1 Introduction

An evaluation was performed to assess the effect of the core-related boron requirements on fluid system design parameters, performance capabilities, and associated plant Technical Specifications and Updated Safety Analysis Report (USAR) commitments. The evaluation was performed using the Westinghouse standard Boration Design Requirements verification process referred to as the BORDER process. BORDER is a controlled process that provides quality assured verification of the continued adequacy of existing system and component designs for each reload, or identifies required design modifications.

6.2.2 Evaluation Overview

Fluid system evaluation of core-related boron requirements using the BORDER code assesses the following fluid system design parameters and performance capabilities:

- The required minimum volume of boric acid in the boric acid storage tank (BAST) and the refueling water storage tank (RWST) to achieve hot and cold shutdown minimum required shutdown margins from an initial at-power operating condition.

- The required minimum volume of boric acid in the BAST and the RWST to borate the RCS to accommodate the positive reactivity insertion due to cooldown from 680°F to 200°F and for RCS mass shrinkage.

- The minimum boration flow capability of the emergency boration flow path to achieve desired RCS shutdown margin in the absence of control rods from an initial at-power operating condition.

- The boron system minimum boration flow capability to compensate for the maximum expected xenon burnout rate at power and loss of shutdown margin conditions.

- The minimum and maximum allowable spray additive tank (SAT) sodium hydroxide concentration to assure that post-LOCA sump pH will be maintained within specified limits.

The BORDER evaluations are based on plant-specific system and component data, licensing bases, and core reload boron requirements. In addition, the BORDER process also reviews and confirms USAR and Technical Specification statements related to RCS boron control and associated fluid system design and performance capabilities.

Reload core boron requirements are evaluated for BOL, MOL, and EOL to ensure that the most limiting boron requirements are evaluated.

6.2.3 Result and Conclusions

Results of the BORDER evaluation show that the KNPP fluid system design bases and performance capabilities are adequate with respect to reload core boron requirements. No Technical Specification changes are required. Several USAR text changes were made as a result of the BORDER analysis. The changes are identified below, and the specific text changes are shown in the USAR revisions included with this submittal:

- Table 9.2-2
  - The description of the nominal 8-percent boric acid solution typically needed to meet cold shutdown requirements was revised.
  - Revisions were made to the minimum rate of boration with one transfer pump and one charging pump at end of life (EOL), the equivalent cooldown rate related to the revised EOL boration rate, the two charging pump boration rate, and the equivalent cooldown rate corresponding to the revised two charging pump boration rate.
- On USAR page 9.2-10, the first sentence in the third paragraph was revised for clarification.

6.3 MECHANICAL ANALYSES

6.3.1 Reactor Internals Structural Analysis

Since the operating parameters for the proposed 422V+ Fuel Upgrade (FU) Program differ from the original design, the reactor vessel system/fuel interface was thoroughly addressed to ensure compatibility and structural integrity of the core during operation. In addition, thermal-hydraulic analyses are required to verify that the core bypass flow limits are not exceeded and to develop pressure drops and upper head temperatures for input to Appendix K emergency core cooling system (ECCS) analyses, non-LOCA accident analyses, and NSSS performance evaluations. The areas most likely to be affected by change in system operating conditions are:

- Reactor internals system thermal/hydraulic performance,
- Rod control cluster assembly (RCCA) scram performance, and
- Reactor internals system structural response and integrity.

6.3.1.1 System Pressure Losses

Total coolant pressure drops across the reactor internals were evaluated for the current plant configuration. These evaluations considered the use of the 422V+ fuel without the intermediate flow mixing (IFMs) devices. This pressure drop data was later used as input to the LOCA and non-LOCA safety analyses.

6.3.1.2 Bypass Flow Analysis

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Analyses were performed to estimate the core bypass flow values and to ensure that the design bypass flow limit for KNPP can be maintained. The results of this assessment confirmed that the design core bypass flow limit of 7.0 percent will be maintained when the 422V+ FU is implemented at KNPP. This result accounts for deleting thimble plugs.

6.3.1.3 Hydraulic Lift Force Analysis

An evaluation was performed to estimate the hydraulic lift forces on the various reactor internal components for the proposed FU/PU Program. This evaluation is required to confirm that the reactor internals will remain seated and stable. The results of this assessment indicated that the overall effect upon the reactor internals of the 422V+ FU/PU to 1772 MWt is negligible compared to the previously analyzed condition. Therefore, it was not necessary to combine lift forces with other forces to calculate the effect on core barrel flange preload.

6.3.1.4 RCCA Scram Performance Evaluation

In general, a plant-specific RCCA scram performance assessment involves the following steps:

1. Obtain actual plant drop time-to-dashpot entry data at no-flow and full-flow conditions for each RCCA location.

2. Develop an analytical model of the plant's driveline configuration and system operating conditions corresponding to those measurements. A driveline is considered to be that subset of components affecting RCCA scram. These components are the fuel, upper core plate, upper and lower guide tubes, upper support plate, reactor closure head penetration, thermal sleeve, control rod drive mechanism (CRDM), rod travel housing, and the RCCA/drive rod assembly. The system operating conditions simply include temperature, pressure, and flow. The analytical model consists of values for parameters that describe geometries of driveline components, component mechanical interaction relationships, hydraulic resistances of flow paths, RCCA/drive rod assembly weight, and system operating conditions.

3. Use a coded algorithm previously developed by Westinghouse with the analytical model to correlate the model to the plant measured drop times. This algorithm, titled DROP, solves Newton's second law of motion. This law states:

$$SF = (W/g) * (dV/dt)$$

where: SF = Sum of various forces acting on the RCCA/drive rod assembly at any time (t)
W = total weight of RCCA/drive rod assembly
g = acceleration due to gravity, 32.2 ft/sec^2
V = assembly velocity, ft/sec
t = drop time after CRDM latch release of drive rod, sec

Correlation involves adjustment of specific code input parameters used to: 1) characterize RCCA drop performance from no (0 percent) flow through full (100 percent) flow based on zero-flow and full-flow core average drop time measurements, 2) isolate and account for the effects of variations in driveline mechanical interference drag force under normal conditions, and variations in driveline flows across the core, based on core-maximum drop time measurements at zero-flow and full-flow respectively.

4. Adjust the model (that is, DROP input parameter values) to account for the new driveline configuration and/or new system operating conditions being considered. Also, conservatively account for:

   a. Component geometric design tolerances
   b. Hydraulic performance uncertainties (related to fuel assembly hydraulic resistance, guide tube/RCCA wear, and reactor coolant flow rate)
   c. Abnormal environmental conditions (particularly seismic events)

5. Assess the impact of such changes in driveline components and/or primary system operating conditions on the limiting RCCA scram characteristics used in the plant accident analyses. These limiting characteristics are the most severe drop time-to-dashpot entry and normalized RCCA scram position-versus-time relationship estimated based on the tolerances, uncertainties, and abnormal environmental conditions identified above.

An analysis was performed to determine the RCCA drop time for the proposed 422V+ FU/PU to 1772 MWt. The maximum RCCA drop time with the seismic allowance was calculated to be 1.59 seconds, which meets the current Technical Specification limit of 1.8 seconds.

6.3.1.5 Structural Evaluation

Structural evaluations are required to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the 422V+ FU. These evaluations were performed for the critical reactor internal components, and the results of these evaluations indicated that the structural integrity of the reactor internals is maintained for the proposed 422V+ FU for KNPP.

6.3.2 Reactor Vessel Structural Evaluation

The KNPP reactor vessel has been evaluated for the structural effects of NSSS operation at conditions of the proposed 422V+ FU Program throughout the duration of the current plant operating license. The evaluations included a review of the Performance Capabilities Working Group Parameters (PCWG) parameters for the KNPP 422V+ FU. The vessel inlet and outlet temperatures were found to be within the bounds of the parameters for the KNPP Steam Generator Replacement (SGR) Program. Therefore, there

are no changes to the operating temperatures or design transients (Reference 72) since they remain conservative for the KNPP FU/PU Program. Additionally, the faulted condition LOCA and seismic loads at the reactor vessel/reactor internals interfaces identified as applicable for the 422V+ fuel upgrade were evaluated. These loads were compared to the allowable faulted conditions loadings which were justified for application to the KNPP reactor vessels. The combined seismic and LOCA loads were found to be substantially less than the allowable faulted condition loads at all three interfaces (main closure, outlet nozzles and core support pads). The conclusion of the evaluations is that the KNPP reactor vessel stress reports as amended for the RSG Program conservatively bound the effects of the 422V+ FU/PU Program. Therefore, operation of the KNPP reactor vessel for the remaining term of the plant's 40 year operating license and for license extension up to 60 years at the conditions defined for the 422V+ FU/PU Program is justified.

For all of the regions of the reactor vessel, the maximum ranges of stress intensity remain below the American Society of Mechanical Engineers (ASME) Code limit of $3S_m$ and the maximum cumulative fatigue usage factors remain less than the ASME Code limit of 1.0.

6.3.3 Design Transients Evaluation

Design transients were reviewed for the SGR Program on the basis of the plant parameters developed for that program. These design transients were reviewed again for the FU Program based on the limiting plant parameters for the 7.4-percent PU condition. A review of the operating conditions for the SGR Program versus those for the FU/7.4-percent Uprating Program is shown in Table 6-3. A comparison of these parameters noted the following:

- T_{hot}: the limiting case is for the high T_{avg} condition; this shows the highest temperature change from no-load to 100-percent power. The change is 606.8°F at full-power to 547°F at no-load (low T_{avg} case shows a smaller change). This change is the same for the uprating as for the SGR Program.

- T_{cold}: the limiting case is for the low T_{avg} condition; this shows the highest temperature change from no-load to 100-percent power. The change is 492.1°F at full-power to 547°F at no-load noted for the SGR Program. If the plant is restricted to a steam pressure no less than 644 psia (saturated pressure at 494°F), then this change is the same for the uprating as for the SGR Program.

- $T\text{-}_{steam}$: the limiting case is for the low T_{avg} condition; this shows the highest temperature change from no-load to 100-percent power. The change is 492.1°F at full-power to 547°F at no-load for the uprating. This change is slightly greater than the 494.0°F at full-power to 547°F at no-load noted for the SGR Program. If the plant is restricted to a steam pressure no less than 644 psia (saturated pressure at 494°F), then this change is the same for the uprating as for the SGR Program.

- Feedwater temperature: The change is 437.1°F at full-power to 32°F (conservative value) at no-load for the FU/PU Program. This change is slightly greater than the 427°F at full-power to 32°F (conservative value) at no-load noted for the SGR Program.

While in general the FU/PU Program results in bounding parameter value changes that are no more severe than the existing design transients developed for the SGR Program, the combined effect of T_{hot}, T_{cold}, steam generator steam temperature, and feedwater pressure exhibited a difference of sufficient magnitude that design transient revisions were felt to be prudent (example : for the high T_{avg} operating condition, while both the SGR and the FU/PU parameters have the same values for T_{hot}, the T_{cold} value is lower by 4.7°F, the steam temperature is lower 6.6°F, and the feedwater temperature is higher by 9.8°F for the FU/PU operating conditions than for the SGR operating conditions). Therefore, all of the design transients that have a power level change during the transient were revised to reflect the FU/PU Program operating conditions.

6.3.4 Reactor Coolant Piping and Supports

6.3.4.1 Introduction

The KNPP FU/PU Program and associated parameters were reviewed for impact on the existing RSG analysis for the following components:

- Reactor coolant loop (RCL) piping
- Primary equipment nozzles
- Primary equipment supports
- The pressurizer surge line piping

The temperature changes associated with the FU/PU program cause potential load changes in the components to be reconciled. The changes in the PCWG temperatures and pressures are factored into the fatigue aspects of the surge line piping evaluation.

6.3.4.2 Input Parameters and Assumptions

The evaluation for the FU/PU Program assumes that all analyses, methods and criteria used in the existing design basis for KNPP will continue to be used.

Three basic sets of input parameters are used in the evaluation of the components identified above for the FU/PU Program.

- PCWG parameters,
- Thermal design transients, and
- Plant life extension to 60 years.

The proposed parameters define the various temperature conditions associated with the potential full power operating conditions of the plant. All of the thermal expansion, seismic, and loss-of-coolant accident (LOCA) analyses performed on the piping systems are done at full power conditions. Thermal design transients and 60 year life extension relate to the fatigue aspects of the analysis. The primary loop piping was designed and analyzed to the USA Standard (USAS) B31.1 Power Piping Code which does not require a formal fatigue analysis. The impact of changes in the proposed temperatures, thermal design transients, and the 60 year life extension were factored into determining the fatigue usage factor and Equation 13 stress for the surge line.

6.3.4.3 Description of Evaluations Performed and Results

The revised PCWG temperatures were assessed to determine the impact on the existing analysis results for the primary reactor coolant loop piping, primary loop nozzles, and primary equipment supports. The assessment considered changes in loads generated as a result of the temperature changes and the potential impact on the listed components. The primary equipment supports were not designed by Westinghouse, but were evaluated based on any measurable changes in load. The evaluation for the FU/PU effort shows no impact on the SGR results.

The evaluation performed to address the effects of the uprating program on the pressurizer surge line stratification analysis focused on the fatigue analysis. The changes in PCWG temperature parameters directly affect fatigue assessments. The difference in temperature between the hot leg and pressurizer is the critical item and was used as a basis for the assessment. The transients and 60-year life did not change from the SGR effort. The evaluation for the FU/PU Program shows that the SGR results are enveloped.

6.3.4.4 Conclusions

The parameters associated with the FU/PU Program for KNPP have been evaluated for impact on the RCL piping, the primary equipment nozzles, the primary equipment supports, and the pressurizer surge line. The evaluation indicates that all components met appropriate allowables. The evaluation for the stated components concluded that the plant FU/PU Program has no adverse effect on the ability of the components to operate until the scheduled end of plant operation. In all cases, the existing evaluations performed for the SGR effort remain applicable.

6.3.5 Primary Loop Leak-before-Break (LBB)

A leak-before-break (LBB) evaluation was performed for the KNPP primary loops to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis for KNPP. The evaluation was documented in WCAP-11411, Rev. 1, and WCAP-15311.

To demonstrate the elimination of RCS primary loop pipe breaks for KNPP, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack which yields a detectable leak rate
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of KNPP
- Demonstrate margin on applied load
- Demonstrate that fatigue crack growth is negligible

These objectives were met as demonstrated in WCAP-11411, Rev. 1, and WCAP-15311.

The LBB evaluations include the applied loads as the input. Both normal operating loads and the faulted loads are used as input to the evaluations.

The effect of the temperature and load changes resulting from the 422V+ Program on the primary loop loads is evaluated.

Based on the evaluation, it was determined that the LBB recommended margins were satisfied. Therefore, the previously provided LBB conclusions will remain unchanged.

In conclusion, an evaluation was performed pertaining to the impact of 422V+ fuel upgrade on the LBB conclusions for the KNPP primary loops. Based on the evaluation, it is determined that the LBB recommended margins were satisfied and that the previous LBB conclusions will remain unchanged.

Table 6-3 Operating Condition Comparison for RTSR/Uprate versus RSG Programs

Parameter	RSG Program		FU/PU Program	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
Plant rated power, MWt	1657.1	1657.1	1780	1780
Loop flow rate, GPM	89,000	89,000	89,000	89,000
Hot leg temperature (T(hot)), °F	606.8	586.3	606.8	590.8
Average loop temperature (T_{avg}), °F	575.3	554.1	573.0	556.3
Cold leg temperature (T(cold)), °F	543.6	521.6	538.9	521.6
No-load temperature, °F	547	547	547	547
Steam flow, x10^6 lb/hr total	7.14	7.11	7.76	7.73
Steam generator steam pressure, psia	791	644	747	634
Steam generator steam temperature, °F	517.0	494.0	510.4	492.1
Feedwater temperature, °F	427.3	427.3	437.1	437.1

Note: Parameters are for limiting 10 percent steam generator tube plugging condition

7 NRC CONDITIONAL REQUIREMENTS FOR THE USE OF 422V+ FUEL

Conditional Requirement 1.

Reference 1 provides acceptance for referencing of topical report WCAP-12610 "VANTAGE + Fuel Assembly Reference Core Report." As stated in Reference 1, it was concluded that WCAP-12610 provides an acceptable basis for the VANTAGE + fuel assembly mechanical design up to a rod-average burnup level of 60 GWD/MTU. The NRC-approval does not include review or approval of higher level burnups as discussed in Appendix B of Reference 2. Similarly, Reference 1 does not address review or approval of the Loss-of-Coolant-Accident (LOCA) analyses methods (Appendix F and G, Reference 3), which are discussed in Reference 4. No other conditional requirements are specified by Reference 1.

1. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'," July 1, 1991.

2. Davidson, S. L., Nuhfer, D. L. (Eds.), "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 and Appendices A through D, June 1990.

3. Kachmar, M. P., Iyengar, J., and Shimeck, D. J., "Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610 Appendices F and G, December 1990.

4. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

Conditional Requirement 2.

Reference 1 provides acceptance for referencing of licensing topical reports WCAP-12610, Appendices F, "LOCA NOTRUMP Evaluation Model ZIRLO™ Modifications," and G, "LOCA Plant Specific Accident Evaluations." As stated in Reference 1, it was concluded that WCAP-12610 Appendices F and G and as clarified in Addendum 4 (Reference 2) is acceptable for referencing in WCAP-12610 licensing applications to the extent specified and under the limitations delineated in the reports and in Reference 1.

1. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

2. Kachmar, M. P., Nissley, M., and Tauche, W., "Additional Information for Appendices F and G of WCAP-12610 Appendix F - LOCA NOTRUMP Evaluation Model ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610, Addendum 4, May 1991.

Conditional Requirement 3.

A limitation as delineated in Reference 1 is as follows:

Although ZIRLO™ is similar to Zircaloy, the criteria of acceptance (10 CFR 50.44, 10 CFR 50.46, and 10 CFR 50, Appendix K) cited in the evaluation are specifically identified as appropriate for Zircaloy-clad fuel. Thus, the staff has concluded that exemptions are needed to allow application of those criteria to ZIRLO™ clad fuel.

This requirement is no longer applicable as a result of recent federal regulation changes (Reference 2), the implementation of ZIRLO™ clad fuel rods is justifiable under 10 CFR 50.59 and requires no prior NRC approval or exemptions.

1. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

2. "Use of Fuel with Zirconium-Based (Other than Zircaloy) Cladding (10 CFR 50.44, 50.46, and Appendix K to Part 50)," Federal Register, Vol. 57, No. 169, Rules and Regulations, pages 39353 and 39355, August 31, 1992.

Conditional Requirement 4.

A limitation as delineated in Reference 1 is as follows:

WCAP-12610, Appendix F, identifies the following changes in the use of the NOTRUMP model to account for ZIRLO™ material properties: clad specific heat, high-temperature creep, rupture temperatures, and circumferential strain following rupture. NOTRUMP/LOCTA-IV retains the methodology given in 10 CFR Part 50, Appendix K, for the treatment of material properties, when prescribed by Appendix K and justified as suitably conservative. The retention of the Baker-Just equation for the calculation of metal/water reaction rate specified in Appendix K is such a case. The staff considered each of these effects as a functional input to the analytical model and found them acceptable in the SER of July 1, 1991 (Reference 2).

1. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

2. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'," July 1, 1991.

Conditional Requirement 5.

A limitation as delineated in Reference 1 is as follows:

In WCAP-12610, Appendix F, Westinghouse identified that the gamma energy distribution methodology (generalized energy distribution model [GEDM]) approved for use in the latest Westinghouse version of WCOBRA/TRAC (staff SER of February 8, 1991) has been incorporated in the NOTRUMP/LOCTA-IV small-break analysis methodology for application to the VANTAGE + fuel with ZIRLO™ material. The staff concludes that the GEDM methodology is applicable to VANTAGE + fuel assemblies with ZIRLO™ material. Therefore, the staff finds that the use of the GEDM in the NOTRUMP/LOCTA-IV small-break methodology to analyze VANTAGE + fuel with ZIRLO™ material acceptable.

1. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

8 REFERENCES

1. "Updated Safety Analysis Report – Kewaunee Nuclear Power Plant," Revision 17 Docket No. 50-305, June 2002 USAR Update.

2. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-NP-A, July 1985.

3. Davidson, S. L., Nuhfer, D. L. (Eds.), "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 and Appendices A through D, June 1990.

4. Burman, D. L., "Appendix E - ZIRLO™ High Temperature Oxidation Tests," WCAP-12610 Appendix E, August 1990.

5. Kachmar, M. P., Iyengar, J., and Shimeck, D. J., "Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610 Appendices F and G, December 1990.

6. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Request for Additional Information on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 1, February 1991.

7. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Second Request for Additional Information on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 2, February 1991.

8. Davidson, S. L., Nuhfer, D. L. (Eds.), "Westinghouse Responses to NRC Additional Issues on WCAP-12610 VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610 Addendum 3, February 1991.

9. Kachmar, M. P., Nissley, M., and Tauche, W., "Additional Information for Appendices F and G of WCAP-12610 Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications; Appendix G - Accident Evaluations LOCA Plant Specific," WCAP-12610 Addendum 4, May 1991.

10. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'," July 1, 1991.

11. Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610, Appendices F, 'Appendix F - LOCA NOTRUMP Evaluation Model: ZIRLO™ Modifications', and G, 'Appendix G - Accident Evaluations LOCA Plant Specific'," October 9, 1992.

12. Brown, U. L., et al., "PERFORMANCE + Fuel Features Generic Safety Evaluation," SECL-92-305, April 5, 1994.

13. Davidson, S. L. (Ed.), et al., "Reference Core Report 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

14. Letter from N. J. Liparulo (Westinghouse) to R. C. Jones (NRC), "WCAP-12610, Appendix B, 'Extended Burnup Fuel Design Methodology and ZIRLO™ Fuel Performance Models,' Addendum 1 [Proprietary]," ET-NRC-92-3732, August 21, 1992.

15. Friedland, A. J. and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A, April 1989.

16. Moomau, W. H., "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology, Kewaunee Nuclear Power Plant (Power Uprate to 1757 MWt-NSSS Power with Feedwater Venturies and 54 F Replacement Steam Generators)," WCAP-15591, Rev. 0, July 2002.

17. Davidson, S. L. (Ed.), et al., "VANTAGE + Fuel Assembly Reference Core Report," WCAP-12610-P-A, April 1995.

18. Davidson, S. L., "Reload Transition Safety Report for Point Beach Units 1 and 2," September 1983.

19. Shimeck, D. J., "1-D Heat Conduction Model for Annular Fuel Pellets," WCAP-14710, September 1996.

20. Davidson, S. L. (Ed.), et al., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary), WCAP-14204-A (Non-Proprietary), October 1994.

21. Letter from Baumann, M. F. (WEPCO) to Document Control Desk (NRC), "14x14, 0.422" OD VANTAGE + (422V+) Fuel Design," NPL 97-0538, November 1997.

22. "Standard Review Plan - 4.2 - Fuel System Design," NUREG-0800, Revision 2, July 1981.

23. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

24. Davidson, S. L. (Ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A (Proprietary), December 1985.

25. Kersting, P. J., et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," WCAP-13589-A, March 1995.

26. Not used.

27. Nguyen, T. Q., et al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

28. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

29. Letter from B. W. Gergos (Westinghouse) to M. Baumann (WEPCO), "14x14 Heavy Fuel Assembly Loss Coefficients," 99WE-G-0025, May 19, 1999.

30. Letter from B. W. Gergos (Westinghouse) to M. Baumann (WEPCO), "FACTS Vibration Test Report - 14 Heavy Fuel Assembly," 99WE-G-0026, May 19, 1999.

31. Letter from B. W. Gergos (Westinghouse) to M. Baumann (WEPCO), "FACTS Hydraulic Test Report - Westinghouse 14x14 Heavy Fuel Assembly," 99WE-G-0023, May 10, 1999.

32. Motley, F. E., Hill, K. W., Cadek, F. F., Shefcheck, J., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A, July 1984.

33. Sung, Y. X., et. al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary), October 1999.

34. Spier, E. M., "Evaluation of Nuclear Hot Channel Factor Uncertainties," WCAP-7308-L-P-A, June 1988.

35. Skaritka, J., "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1, July 1979.

36. Letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2515, October 9, 1981; and letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2572, March 16, 1982.

37. Letter from Berlinger, C. (NRC) to Rahe, E. P., Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

38. Schueren, P. and McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalties," WCAP-11837, May 1988.

39. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), "Acceptance for Referencing of Licensing Topical Report WCAP-11837," October 1989.

40. Not used.

41. Not used.

42. Not used.

43. Not used.

44. Foster, J. P., Sidener, S, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A with errata, July 2000.

45. Hargrove, H. G., "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

46. Hugel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882-P-A, April 1999.

47. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984.

48. Liu, Y. S., et al., "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

49. Risher, D. H., Jr. and Barry, R. F., "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-P-A, June 1975.

50. Sung, Y. X., et al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A, October 1999.

51. "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis," WCAP 12945-P-A, (Proprietary) Volume 1 (Revision 2) and Volumes II-IV (Revision 1), and WCAP-14747 (Non-Proprietary).

52. Letter, R. C. Jones (NRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945 (P), Westinghouse Code Qualification Document for Best Estimate Loss-of-Coolant Analysis," June 28, 1996.

53. "Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," WCAP-14449-P-A, Revision 1, October 1999.

54. Letter, C. A. Carpenter (NRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-14449 (P) 'Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection," (TAC NO. M94035), May 1999.

55. Meyer, P. E., "NOTRUMP – A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, August 1985.

56. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A, August 1985.

57. Thompson, C. M., et al., "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," WCAP-10054-P-A, Addendum 2, Rev. 1, July 1997.

58. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301, June 1974.

59. "Reference Core Report Vantage 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

60. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model Summary," WCAP-8339-NP-A, July 1974.

61. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model: Supplementary Information," WCAP-8471-P-A (Proprietary) and WCAP-8472-NP-A (Non-Proprietary), April 1975.

62. Westinghouse Technical Bulletin, "Post-LOCA Long-Term Cooling: Boron Requirements." NISD-TB-86-08, October 31, 1986.

63. "Kewaunee Nuclear Power Plant, Revision to the Design Basis Radiological Analysis Accident Source Term," Document Number NRC-02-024, March 19, 2002.

64. PD2-01-46, Rev. 1 (01WP-S-002), M. L. Lewis, M. A. Marzean and A. Lahlou, "Kewaunee Fuel Transition Work Report, Revision 1 to Fuel Assembly Compatibility Report for the Supply of 14x14 Westinghouse 422V+ Fuel Assembly," November 1, 2001.

65. O'Donnell P. F., "Storage of Westinghouse 422V+ Fuel Assemblies in the Kewaunee Spent Fuel Storage Racks and Fresh Fuel Storage Vault," LTR-WFE-02-183, July 12, 2002.

66. SER related to Amendment No. 193 to Facility Operating License No. DPR-24 and Amendment No. 198 to Facility Operating License No. DPR-27 WEP Point Beach NP Units 1 & 2, Docket Nos. 50-266 and 50-301.

67. "Technical Specifications - Kewaunee Nuclear Power Plant," Docket No. 50-305, Amendment 159, October 2001.

68. WCAP-12488, Addendum 2, "Revisions to Design Criteria," April 2002.

69. Letter PD-02-153, Rev. 1 "Seismic/LOCA Analysis Summary Report of Kewaunee 14x14 442V+ Implementation," June 17, 2002.

70. Letter from R. O. Owoc (Westinghouse) to J. Jensen (KNPP), "Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, Kewaunee RSG – Final Licensing Report Submittal," KEW-LIC-00-096/LTR-ESI-00-302, November 29, 2000.

71. NRC-02024, Kewaunee Nuclear Power Plant, Revision to the Design Basis Radiological Analysis Accident Source Term, March 19, 2002.

72. Abbott, S. L., "Addendum to the Stress Reports for the Point Beach Nos. 1 and 2 Reactor Vessels (RSG/Uprating Evaluation)," WCAP-14448, August 1995.