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SUBJECT: Provides supplemental info in response to 980826 telcon with NRC.Non-proprietary Rev 1 to EMF-2095(NP) & proprietary Rev 1 to EMF-2095(P) repts encl.Proprietary rept withheld,per 10CFR2.790.

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Public Service

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September 28, 1998

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

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Proposed Amendment 152 to the Kewaunee Nuclear Power Plant Technical Specifications - Supplemental Information

References:
1) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated April 15, 1998.
2) Letter from M. L. Marchi (WPSC) to Document Control Desk (NRC) dated August 13, 1998.

On April 15, 1998, Wisconsin Public Service Corporation (WPSC) submitted proposed Technical Specification (TS) amendment 152 to implement the improvements realized by a modified fuel design and to reflect changes to the Kewaunee plant operating conditions (Reference 1). Reference 2 provided the WPSC response to a NRC request for additional information. Attachment 1 to this letter provides supplemental information in response to a telephone conference with the NRC on August 26, 1998 and subsequent conversations with the NRC Project Manager.

In addition, Attachment 2 provides a proprietary copy of Siemens Power Corporation report EMF-2095(P) Revision 1, "Mechanical Design Evaluation for Kewaunee Cycle 23, Reloads KEW-19, KEW-18 and KEW-17 Fuel Assemblies," with the accompanying affidavit attesting to the proprietary nature of the information. Accordingly pursuant to 10 CFR 2.790, WPSC requests that the information be withheld from public disclosure. Attachment 3 provides the associated non-proprietary report.

Please contact Rick Pulec of my staff at (920) 388-8376, should you have any further questions or require additional information.

Sincerely,

M Marchi

Mark L. Marchi
Site Vice President-Kewaunee Plant

RPP/jmf
Attach.

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ATTACHMENT 1

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

September 28, 1998

Supplemental Information Related
to
Proposed Technical Specification Amendment 152

This attachment provides supplemental information requested by the NRC staff during a telecon on August 26, 1998 and subsequent conversations with the NRC Project Manager.

1) **Compliance with Topical Report Conditions - The response to NRC question 1 lacks a discussion of the methods used for mechanical and nuclear designs of the reload fuel (the SPC heavy fuel) and address the compliance with the limitations imposed on the approved methods.**

The mechanical design was performed by Siemen's Power Corporation (SPC) using SPC's NRC-approved (References 1, 2, 15, and 16) mechanical desigu methodology. SPC has reviewed the conditions imposed by the Safety Evaluation Reports for the approved mechanical desigu methodology and assured WPSC that the Kewaunee analyses is in conformance with these conditions.

The nuclear design was performed by WPSC using WPSC's NRC-approved (Reference 3) nuclear design methodology. The SER for Reference 3 specifically limits WPSC to the use of codes in the EPRI ARMP code package. WPSC continues to use the ARMP package in the manner approved by the NRC staff along with the latest code enhancements. WPSC has also recently begun using the MICBURN code for the neutronic analysis of the Gadolinia bearing Cycle 23 fuel. MICBURN is part of the ARMP package and is used to obtain Gadolinia cross sections that are input to WPSC's cross section generator CPM2. The use of MICBURN for Cycle 23 has been evaluated under 50.59 as part of the Cycle 23 Reload Safety Evaluation (RSE) report (Reference 7).

Fuel desigu information needed by WPSC to perform the nuclear desigu is obtained directly from the fuel vendor, SPC. The fuel contract defines reload core design and analysis interactions with the fuel vendor. The schedule for the interactions is also defined. Among other things the interactions allow WPSC to establish the reload core design, receive a list of potential licensing issues from the vendor, receive fuel drawings and specifications for the reload, and receive a reload specific mechanical design report. WPSC is also provided with any and all fuel desigu data that is required by WPSC to perform its reload analysis responsibilities.

Design changes resulting from fuel vendor design or design methods changes are identified via a WPSC nuclear fuel design change review process or as a result of the audits that WPSC conducts of the fuel vendor. The fuel assembly design change review process requires a complete evaluation of the design changes with respect to 10CFR50.59.

WPSC is responsible for the reload core design and safety evaluation. Fuel design data provided by the fuel vendor is used in these processes. The fuel design data that is provided includes geometric data for the fuel assembly and fuel rod, thermal hydraulic data, neutronic data, and materials data. In addition WPSC has received the DNBR correlation that is applicable to the SPC fuel design plus all of the supporting measured data used to support this DNBR correlation. WPSC uses the SPC provided fuel design data to create inputs to the WPSC computer models that are needed for the reload core design and safety analysis processes.

Procedures established by the Nuclear Fuels group provide a comprehensive set of design controls and review processes for the reload core design and safety evaluation activities. The procedures insure that activities related to reload core design and safety analysis are performed in accordance with the WPSC Operational Quality Assurance Program and hence Appendix B of 10CFR50. The integrity of the fuel design data and the integrity of the resulting controlled design inputs to the WPSC computer models are also assured by these WPSC internal procedures.

Fuel design data required for Loss of Coolant Accident (LOCA) analyses, currently performed by Westinghouse, is provided to Westinghouse under a tri-partite proprietary information agreement. The fuel design data that has been provided to Westinghouse for LOCA purposes includes most of the data that WPSC requires for reload analyses plus additional data specific to the LOCA analysis processes and models.

Have you verified that each and all of the conditions imposed on the methods nsed for large and small break LOCA analyses were met?

WPSC has completed a review of NRC safety evaluations for the LOCA analyses methods to verify that conditions imposed have been satisfied with the results provided on the tables that follow on the subsequent pages. WPSC's review has concluded that all conditions imposed in the applicable NRC SERs have been satisfied.

Small Break LOCA - WCAP-14103, "Westinghonse Small Break Loss-of-Coolant Accident Kewaunee NOTRUMP Analysis," dated June 1994

Analysis Topical Reports	NRC Safety Evaluation/Conditions for Use	Disposition
WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," dated August 1985.	Letter from C. O. Thomas (NRC) to E. P. Rahe Jr. (Westinghouse) dated May 21, 1985/ No conditions specified.	Not Applicable.
WCAP-10079-P-A, 'NOTRUMP, A Nodal Transient Small Break and General Network Code," dated August 1985.	Letter from C. O. Thomas (NRC) to E. P. Rahe Jr. (Westinghouse) dated May 23, 1985/ No conditions specified.	Not Applicable.
WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analysis," dated June 1974.	Letter from D. B. Vassallo (NRC) to C. Eicheldinger (Westinghouse) dated May 30, 1975/ Conditions specified:	
	1. Typical, current Westinghouse two, three, and four loop plants.	1. Kewaunee is a typical, Westinghouse two loop plant.
	2. Plants with dry, subatmospheric, or ice containments.	2. Kewaunee has a dry containment.
	3. Plants with power ratings up to 3800 Mwt.	3. Kewaunee has a power rating of 1650 Mwt.
	4. Plants only utilizing bottom flooding emergency core cooling systems.	4. For SBLOCAs, Kewaunee relies on high head safety injection to the bottom of the reactor vessel via the RCS cold legs and the vessel downcomer.

Large Break LOCA - Transmittal from Westingbouse to WPSC dated June 18, 1998 (WPS-98-20)

Analysis Topical Reports	NRC Safety Evaluation/Conditions for Use	Disposition
WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology," Vol.1, Rev. 1, including Addenda, dated December 1988	Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse) dated August 29, 1988/ Conditions specified:	Not applicable.
	1. Exemptions will be needed to Appendix K Items 1.D.3 and 1.D.5 regarding core exit liquid carryover fraction and refill/reflood heat transfer, respectively.	NRC granted required exemption to WPSC on November 19, 1996.
	2. Sensitivity studies should be performed in each plant-specific licensing calculation to determine the location under which the hot assembly is placed to obtain the highest PCT.	WPS-98-20 identifies that a sensitivity study identified an interior assembly under a support column surrounded by non-guide tube assemblies as the hot assembly for highest PCT.
	3. Any relaxation of the assumption of the instantaneous equilibrium initial condition in the decay heat calculation using 1979 ANS Standard will need staff review and approval.	WPS-98-20 identifies the use of the 1979 ANS Decay Heat Model for the calculation.
Including Addenda 4 approved February 1991	Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse) dated February 8, 1991/ No conditions specified.	Not applicable.

Analysis Topical Reports	NRC Safety Evaluation/Conditions for Use	Disposition
WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped Upper Plenum Injection," Vol. 2, Rev. 2, and Addenda dated December 1988	Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse) dated August 29, 1988/ Conditions specified are same for WCAP-19024-P-A, Vol.1.	Same as for Vol. 1.
WCAP-8327, "Containment Pressure Analysis Code (COCO) dated June 1974.	Letter from D. B. Vassallo (NRC) to C. Eicheldinger (Westinghouse) dated May 30, 1975/ Conditions specified are same as for SBLOCA.	Resolution same as for SBLOCA except the bottom flooding ECCS. The NRC approved this code as part of a larger ECCS evaluation model. However, this is a containment response model, and not relevant to the core reflood section of the model. Therefore, core fiooding requirements for its use are not applicable.
WCAP-10851-P-A, "Improvcd Fuel Performance Models for Westinghouse Fuel Rod Desigu and Safety Evaluations," dated August 1988.	Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse) dated May 9, 1988/ Conditions specified: Fuel rods with nominal fabricated fuel-cladding gap sizes of 10 mils or less.	Kewaunee fuel has a fuel-cladding gap of less than 10 mils.

2) Acceptability of the SPC Heavy Fuel Design - The SPC heavy fuel assemblies have design featnres different from the fuel in the existing core. You have not addressed the impact of the fnel design changes on the mechanical, nuclear and thermal-hydraulic designs and justified that the new fuel design is acceptable for the Cycle 23 reload application by showing that the applicable design criteria are met.

SPC heavy assemblies have been in the core during Cycles 21 and 22 as lead test assemblies. The mechanical desigu of the lead test SPC heavy assemblies was shown to meet applicable design criteria in Reference 4. The nuclear and thermal-hydraulic designs were shown to meet applicable design criteria in Reference 5. A similar approach has been be used to verify that applicable design criteria are met for the full batch of SPC heavy feed assemblies going into Cycle 23.

Reference 6 (included as Attachment 2) shows mechanical desigu criteria applicable to the SPC heavy fuel in Cycle 23 are met. Table 3.2 (Page 3-6) of Reference 6 lists each mechanical design criterion and confirms the Cycle 23 design meets that criterion.

Reference 7 (transmitted under separate cover letter) shows nuclear and thermal hydraulic design criteria applicable to the SPC heavy fuel in Cycle 23 are met. Tables 3.1.1 through 3.16.1 of Reference 7 compare Cycle 23 important safety parameter results to the current safety analysis, demonstrating that the Cycle 23 design is more conservative than results of the current safety analysis.

Page 2 of attachment to the September 5, 1997 letter stated that "... the primary impact of the (new) fnel rod design changes is an increase in the end of life internal rod pressnre." What is the basis for the above statement?

The statement was based on the Reference 4 results for the SPC heavy lead test assemblies used in Cycle 22 as compared to the SPC standard assembly results in Reference 8. Of all the design criteria examined, the internal rod pressure showed the most marked change when comparing SPC heavy fuel to SPC standard fuel. This is primarily due to the smaller plenum volume resulting from the larger plenum spring volume and to power history differences. However, as for all other design limits, Table 3.2 of Reference 6 shows that the design limit for internal rod pressure is met for the SPC heavy fuel in Cycle 23.

Page 2 for the docnment referenced above indicated that the design limit for tbe internal rod pressure is 3050 psia. What is the basis for the acceptance criterion?

As shown on Page 3-8 of Reference 6 (Table 3.2, Criteria Section 3.3.7), the internal rod pressure design limit value of 3050 psia is obtained by taking the Kewaunee RCS pressure of 2250 psia and adding 800 psia. The basis for the internal rod pressure design criterion is contained in Supplement 5 of Reference 1 and was accepted by the NRC on page 8 of the incorporated SER.

The results of the impact evaluation for the uew fuel design should be provided for the staff to revicw before the staff completion of the TS change review.

The SPC mechanical design report (Reference 6) and the Cycle 23 RSE (Reference 7) are provided as Attachment 2 of this letter and under separate cover letter, respectively.

What fuel burnup limits are applied to the SPC hcavy fuel?

The SPC NRC-approved mechanical design methodology (References 1, 2, 15, and 16) establishes a 62 GWD/MTU peak rod fuel burnup limit. Section 2 of Reference 6 verifies that this peak rod fuel burnup limit applies to the SPC heavy fuel in Cycle 23.

3) LOCA Analyses - The response to NRC question 3 lacks discussion of:

Specific breaks that were analyzed to determiue the limiting cases for both large and small break LOCAs.

Large Break LOCA: Westinghouse performs the Large Break LOCA calculations for the Kewaunee plant. The Reference 9 Large Break LOCA methodology WCAP includes the results of Large Break LOCA break size spectrum calculations performed for a generic plant. Section 6-1-4 on Pages 6-8 through 6-12 discusses break size spectrum realistic calculations performed at bounding conditions for a generic plant. Section 7-4-2 on Pages 7-66 through 7-69 discusses some additional break size spectrum Appendix K calculations performed for a generic plant.

The table below summarizes the Large Break LOCA break spectrum generic plant sensitivity cases run by Westinghouse using information from pages 6-34 (Table 6-3), 7-67, and 7-68 of Reference 9.

Large Break LOCA Generic Plant Break Size Spectrum Case Summary
(Information taken from WCAP-10924-P-A, Revision 2 (Reference 9))

Generic Plant Break Description	Generic Plant Model Type	Generic Plant Blowdown PCT (°F)	Generic Plant Refiood PCT (°F)	WCAP-10924-P-A, Revision 2 (Reference 9) Page Number
0.4 DECLG	Appendix K	1907	2052	7-67
0.6 DECLG	Appendix K	1623	2004	7-68
--	--	--	--	--
0.6 DECLG	Realistic	1678	1840	6-34
0.8 DECLG	Realistic	1336	1385	6-34

Generic Plant Break Description	Generic Plant Model Type	Generic Plant Blowdown PCT (°F)	Generic Plant Reflood PCT (°F)	WCAP-10924-P-A, Revision 2 (Reference 9) Page Number
1.0 DECLG	Realistic	677	506	6-34
--	--	--	--	--
0.1 CLS	Realistic	678	NSCH	6-34
0.4 CLS	Realistic	1551	1463	6-34
0.6 CLS	Realistic	1728	1400	6-34
0.8 CLS	Realistic	1659	NSCH	6-34
1.0 CLS	Realistic	1485	NSCH	6-34
1.2 CLS	Realistic	1401	949	6-34
--	--	--	--	--
1.0 DEHLG	Realistic	942	399	6-34

NOTES:

1) The cases in this table were run with a generic plant and were used to establish the limiting break for two loop UPI plants on a generic basis. The results of the generic cases were used to establish the 0.4 DECLG break as the limiting break for Appendix K calculations and the 0.6 DECLG break as the limiting break for Superbounded calculations

2) x.x DECLG is a Double Ended Cold Leg Guillotine break with a discharge coefficient of x.x.

3) x.x CLS is a Cold Leg Split break with a break area equivalent to x.x times the cross seetional area of the cold leg.

4) x.x DEHLG is a Double Ended Hot Leg Guillotine break with a discharge coefficient of x.x.

5) NSCH means that there was No Significant Clad Heatup during the reflood portion of the case in question.

These generic plant calculations established the 0.4 DECLG break as the most limiting break for Appendix K two loop UPl Large Break LOCA calculations and the 0.6 DECLG break as the most limiting for Superbounded two loop UPI Large Break LOCA calculations. Based on the generic plant results, Westinghouse used a 0.4 DECLG break for the Kewaunee Appendix K case and a 0.6 DECLG break for the Kewaunee Superbounded case. The table below shows the Reference 10 Kewaunee Large Break LOCA PCT results for SPC heavy fuel.

Kewaunee Large Break LOCA Results
(Information taken from Westinghouse letter WPS-98-020 (Reference 10))

Break Description	Model Type	Blowdown PCT (°F)	Reflood PCT (°F)
0.4 DECLG	Appendix K	1670	1872
0.6 DECLG	Superbounded	1823	1812

NOTES:

1) The cases in this table were run with the Kewaunee specific model using the limiting break sizes established by the generic plant cases shown in the previous table.

2) x.x DECLG is a Double Ended Cold Leg Guillotine break with a discharge coefficient of x.x.

Small Break Sizes: Westinghouse performs the Small Break LOCA calculations for the Kewaunee plant. For Small Break LOCA, Westinghouse performed plant specific break size spectrum calculations using the Kewaunee plant model. The Reload Safety Evaluation uses information from page 16 (Table 4) of Reference 11 to summarize Kewaunee specific Small Break LOCA break size spectrum cases run by Westinghouse.

Kewaunee Specific Small Break LOCA Break Size Spectrum Case Summary
(Information taken from Page 16 of WCAP-14103 (Reference 11))

Break Size (diameter, inches)	PCT (°F)
2	NCU
3	1053
4	871
6	NCU

NOTES:

1) The above cases were run with the Kewaunee specific Small Break LOCA model. They assumed 25% steam generator tube plugging and SPC standard fuel. The results of the cases support a limiting break size of 3 inches for Kewaunee. Later evaluations based on these cases were performed to increase the steam generator tube plugging to 30% (Reference 17) and to include SPC heavy fuel (Reference 12). These evaluations resulted in a net PCT benefit. Therefore the evaluated PCT with SPC heavy fuel, 30% steam generator tube plugging, and a 3 inch diameter break is 1041°F. An additional 109°F PCT benefit may be taken once all of the SPC standard fuel in the core becomes twice burned. Cycle 23 contains some once burned SPC standard fuel, so the PCT is 1041°F for Cycle 23.

2) NCU means that there was No Core Uncovery for the case in question and thus no hot spot clad temperature.

Based on these results, Westinghouse uses a three inch break for Kewaunee Small Break LOCA analyses and evaluations. Reference 12, which is an evaluation based on Reference 11, establishes the Kewaunee Small Break LOCA PCT at 1041°F for SPC heavy fuel until such time that all Siemens standard fuel also in the core is twice burned. After that time, a PCT benefit of 109°F may be taken. Since Cycle 23 contains some once burned SPC standard fuel, the PCT of 1041°F applies.

List the Evaluation Model Updates (for small break LOCA aualysis - Attachment 4, Page 2 of April 15, 1998 letter) aud the justificatiou for acceptance of the EM updates.

The Evaluation Model Updates fall into two basic categories: changes initiated by WPSC to reflect operational changes and changes initiated by Westinghouse to correct code modeling errors or make model improvements. The former includes items like increases in steam generator tube plugging, reduced reactor coolant flow, and fuel design changes. Tabulated below is a listing of the current model updates for small break LOCA, the justification for acceptance by WPSC, and as applicable, the Westinghouse letter that notified the NRC of the model update.

Model Update	Basis for Acceptance	Westinghouse Ltr. #
Safety injection to broken loop	Revised model assumption is more limiting and results in increased peak clad temperature	NTD-NRC-94-4130
Improved loop condensation model	Prototypical tests at COSI facility	NTD-NRC-94-4130
Error in friction factor values in SPADES code which provides inputs to NOTRUMP	Correction of code error	NTD-NRC-95-4409
Potential containment spray actuation results in earlier transfer to recirculation with SI interruption or reduction	Credible event sequence	NTD-NRC-95-4409
SBLOCTA code - resolve deficiency with inadequate nodalization detail, correct fluid conservation equation, and incorporate revised fuel rod internal pressure model	Correction of code deficiency/error and adoption of approved model change.	NTD-NRC-95-4409
NOTRUMP coding typographical error for specific enthalpy	Correction of code error	NSD-NRC-96-4639

Model Update	Basis for Acceptance	Westinghouse Ltr. #
Library routine - improper specification of double precision variables	Correction of code error	NSD-NRC-96-4639
SBLOCTA code - error in fuel rod initialization adjustment	Correction of code error	NSD-NRC-98-5575
Fuel rod clad creep and strain model - logic errors	Correction of code error	NSD-NRC-98-5575
Increase steam generator tube plugging to 30% and reduced RCS flow to 83,400 gpm/loop	WPSC requested change	Not Applicable
Change to SPC Heavy fuel	WPSC requested change	Not Applicable

WPSC is also aware that Northern States Power Corporation has recently completed a baseline re-evaluation of the small break LOCA analysis with a resulting larger increase in PCT than that previously estimated from the cumulative model changes. This was reported to the NRC in Reference 18. Like the NSP analysis, the WPSC SBLOCA analysis has substantial margin to the 2200 degrees Fahrenheit acceptance criterion, and therefore, this is not an immediate safety concern. WPSC will however be monitoring the results of the Westinghouse investigation and resolution of this anomaly. In addition, Westinghouse is in the process of performing a new small break LOCA analysis to support the steam generator replacement scheduled for Spring 2000.

4) What is the basis for allowing 40% of the fuel rods to have an FΔHN value exceeding the FΔHN at which DNB occurs during a locked rotor event?

The acceptance criteria for this specific safety analysis parameter of the Locked Rotor Accident is described in section 3.12.2 (page 3-74) of Reference 13. The acceptance criterion is that the number of fuel rods expected to experience a DNBR of less than 1.30 should not exceed the number of fuel rods required to fail in order to yield doses due to released activity which will exceed the limits of 10CFR100.

In section 3.12.5 (page 3-77) of Reference 13 the safety analysis value for this parameter is explicitly identified as 40%. The reload safety evaluation must therefore show that the cycle specific calculated value for this parameter is less than the safety analysis criterion of 40%.

NRC has approved these reload safety evaluation methods for the evaluation of the Locked Rotor Accident and the accident specific acceptance criterion of 40% used for this safety analysis parameter (see sections 2.4 and 2.7 of Reference 14).

Additionally from the safety analysis of the locked rotor transient, it is determined that the initial heavy fuel rod $F\Delta H$ that results in a heavy fuel rod going below the MDNBR limit during the event is 1.513. The initial standard fuel rod $F\Delta H$ that results in a standard fuel rod going below the MDNBR limit is 1.420. Cycle 23 Reload Safety Evaluation analyses have recently been completed. These analyses indicate that the percentage of fuel rods that will experience DNB, i.e., the heavy fuel rods that have $F\Delta H \geq 1.513$ and the standard fuel rods that have $F\Delta H \geq 1.420$, during a locked rotor event is 9.2% for the heavy fuel rod and 0.0% for the standard fuel rod.

The percentage of fuel rods that would experience DNB in the Cycle 23 reload core is well below the Kewaunee locked rotor acceptance criteria of 40%.

5) Discuss how mixed cores were evaluated for SBLOCA and LBLOCA.

SMALL BREAK LOCA: Westinghouse performed an assessment of the SPC heavy fuel based on the existing SBLOCA SPC standard fuel analysis using the Kewaunee specific small break LOCA NOTRUMP model. There is no SBLOCA transition core effect due to hydraulic differences between the two assembly types, since the clad outer diameters are the same in the SPC standard and heavy assemblies, the grid spacers are located at the same elevations, and the losses throughout the core are approximately the same. Therefore, only the effect of the differences in initial fuel temperatures, rod internal pressures, and fuel dimensions must be considered in evaluating the transition core effects. Thus, only the NOTRUMP hydraulic transient for SPC standard fuel was used to perform a revised rod heatup calculation for SPC heavy fuel. The calculation showed that the SPC heavy fuel provides a benefit and increases the margin to the regulatory PCT limit. Reference 12 documents the results of the assessment.

LARGE BREAK LOCA: Westinghouse evaluated the transition from SPC standard to SPC heavy fuel using the Superbounded model. The evaluation is documented in Reference 10. There is no LBLOCA transition core effect due to hydraulic differences between the two assembly types, since the clad outer diameters are the same in the SPC standard and heavy assemblies, the grid spacers are located at the same elevations, and the losses throughout the core are approximately the same. Therefore, only the effect of the differences in initial fuel temperatures, rod internal pressures, and fuel dimensions must be considered in evaluating the transition core effects.

To determine the effect of the initial fuel temperatures, rod internal pressures, and other fuel characteristics, a Superbounded calculation was performed in which the hot assembly is modeled as a fresh assembly of SPC heavy fuel, and the remainder of the assemblies are modeled as once burned (or more) SPC standard fuel. This case showed less severe results in the limiting reflood portion of the event than the equivalent case with all SPC standard fuel.

Since the Kewaunee LBLOCA results with a full core of SPC heavy fuel are less limiting than the Kewaunee LBLOCA results with a full core of SPC standard fuel, consideration was also given to the instance where the hot assembly is a once burned SPC standard fuel assembly, with a minimnm assembly burnup of 8000 MWD/MTU and a minimum fuel rod burnup of 3500 MWD/MTU. The results of this case were less limiting than the case where the hot assembly was a fresh SPC heavy assembly.

The transition core analysis therefore bounds all possible loading patterns during the transition from SPC standard to SPC heavy fuel.

6) Questions concerning the specific proposed TS changes:

Several of the added corrective actions to this TS are not consistent with the Westinghouse standard TS (STS) as stated by the licensee.

WPSC agrees. The proposed actions were based upon existing Kewaunee requirements restructured to match the STS granted to the Ginna Plant. WPSC is withdrawing this scope of the proposed change..

The proposed TS 3.10.u removes the minimum DNBR value. This value should be included in the specification or Basis.

WPSC will include in the TS Bases section the minimum values of 1.14 for the HTP correlation and 1.3 for the W-3 correlation with an explanation for conditions of use of these two correlations.

In TS 3.10.k, the average core temperature is 568.8 F, while the value for the core average temperature used in the transient analysis is 573.1 F. The licensee states on page 6 of Attachment 1, that "The design basis safety analyses have been re-analyzed and/or evaluated at the RCS average temperature."

The higher transient analysis temperature includes instrument uncertainty and reflects operation at 102% power, both conservative adjustments for minimum DNBR evaluations. For those transients where a lower temperature is more limiting the instrument uncertainty is used to reduce analysis assumption for average core temperature.

In TS 3.10.l, the minimum RCS is specified at 2205 psig while the value used for the analysis is 2185 psig. Specifically, the note on the page that lists the assumptions states that the input parameters with an asterisk were changed from the previous evaluation. The RCS pressure is a parameter with an asterisk. Nevertheless, the licensee states in the justification for the TS changes, that "the limit for RCS pressure is not changed in the analysis or in the Technical Specifications."

The lower transient analysis pressure includes instrument uncertainty, a conservative adjustment for minimum DNBR evaluations. For those transients where a higher pressure is more limiting the uncertainty is used to increase the RCS pressure. WPSC agrees the last statement is incorrect; it should have only identified the limit for RCS pressure as not changed in the Technical Specifications. This will be corrected in Attachment 1 of the proposed amendment.

TS 3.10.m proposed to reduce the miuimum RCS flow from 89,000 to 85,500 gpm per loop. This change is not conservative as compared with the values used in the aualyses: For the LOCA analysis, the flow rate was assumed at 83,400 gpm/loop. For the transient analysis, the flow rate was assumed at 83,500 gpm/loop.

The proposed TS discussion described analysis changes particularly when the changes were less conservative for a particular transient. For those transients where minimum DNBR is a concern, the analyses provides conservative results with the lower assumed RCS flow values which include instrument uncertainty and bound the TS proposed value. However for transients where a higher RCS flow provides a more limiting analysis result, the flow value was not adjusted. One example is the main steam line break analysis where the RCS flow assnmed equates to 0% steam generator plugging to ensure a bounding analysis. For the steam demand increase transient (excessive load increase) minimnm RCS flow is used since this is conservative for the DNBR assessment of this transient. Analysis results of the excessive load increase transient at lower RCS flow were not significantly changed from the higher RCS flow results. Both the low and high RCS flow analyses for this transient have large margins to the acceptance criteria.

A revision to the proposed amendment to reflect the above described changes will be transmitted under separate cover and identified as Proposed Amendment 152a.

REFERENCES:

1) Qualification of Exxon Nuclear Fuel for Extended Burnup, XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 & 5, Exxon Nuclear Company, Richland, WA 99352, October 1986.

2) Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnup to 62 GWd/MTU, ANF-88-133(P)(A) and Supplement 1, Advanced Nuclear Fuels Corporation, Richland, WA 99352, December 1991.

3) Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, topical report WPSRSEM-NP-A entitled, "Qualification of Reactor Physics Methods for Application to Kewaunee, " dated September 29, 1978.

4) Mechanical Design Evaluation for Kewaunee Reload KEW-18 Lead Fuel Assemblies, EMF-96-127 Revision 1, Siemens Power Corporation Nuclear Division, Richland, WA 99352, September 1996.

5) Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, report entitled "Reload Safety Evaluation Cycle 22," dated August 1996.

6) Mechanical Design Evaluation for Kewaunee Cycle 23, Reloads KEW-19, KEW-18, and KEW-17 Fuel Assemblies, EMF-2095 (P&NP) Revision 1, Siemens Power Corporation Nuclear Division, Richland, WA 99352, September 1998.

7) Wisconsin Public Service Corporation, Kewaunee Nuclear Power Plant, report entitled "Reload Safety Evaluation Cycle 23," dated September 1998.

8) Mechanical Design Evaluation for Kewaunee Reload KEW-18 Standard Fuel Assemblies, EMF-96-126 Revision 1, Siemens Power Corporation Nuclear Division, Richland, WA 99352, September 1996.

9) WCAP-10924-P-A, Volume 2, Revision 2, and Addendum, Dederer, S.I., et al., Westinghouse Large-Break LOCA Best-Estimate Methodology, December 1988, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection; Addendnm 1: Responses to NRC Questions.

10) "Kewaunee Final Large Break LOCA USAR Updates for Siemens 14x14 Heavy Fuel," Westinghouse letter WPS-98-020 from Stephen P. Swigart to Dave Wanner, dated June 18, 1998.

11) WCAP-14103, Capone, S.J., et al., Westinghouse Small-Break Loss-of-Coolant Accident Kewaunee NOTRUMP Analysis, June 1994.

12) "Safety Assessment for Transition to Siemens 14x14 Heavy Fuel," Westinghouse letter WPS-97-503 from J.A. Bugica, Jr. to D. Wanner, dated February 10, 1997.

13) Kewaunee Nuclear Power Plant, topical report WPSRSEM-NP-A, Revision 2 entitled, "Reload Safety Evaluation Methods for Application to Kewaunee, " dated October 1988.

14) "Wisconsin Public Service Corporation "Reload Safety Evaluation Methods for Application to Kewaunee (TAC No. 65155)," letter from J.G. Giitter (NRC) to D.C. Hintz (WPSC) dated April 11,1988.

15) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1984.

16) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 3 and 4, Advanced Nuclear Fuels Corporation, Richland, WA 99352, June 1990.

17) "Safety Assessment for Increased SGTP to 30%," Westinghouse letter WPS-96-521 from J.A. Bugica, Jr. to D. Wanner, dated November 4, 1996.

18) "Report of Corrections to ECCS Evaluation Models," Northern States Power Corporation to Nuclear Regulatory Commission dated September 8, 1998.

ATTACHMENT 3

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

September 28, 1998

EMF-2095(NP) Revision 1, "Mechanical Design Evaluation for Kewaunee Cycle 23, Reloads KEW-19, KEW-18 and KEW-17 Fuel Assemblies" Siemens Power Corporation, September 1998