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Public Service

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August 13, 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
Response to Request for Additional Information - Proposed Amendment 152
To the Kewaunee Nuclear Power Plant Technical Specifications

References: 1) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated April 15, 1998.

2) Letter from W. O. Long (NRC) to M. L. Marchi (WPSC) dated July 1, 1998.

On April 15, 1998, Wisconsin Public Service Corporation (WPSC) submitted a proposed Technical Specification (TS) amendment to implement the improvements realized by a modified fuel design and to reflect changes to the Kewaunee plant operating conditions (Reference 1). In order to complete review of the proposed TS amendment, the NRC staff has requested additional information (RAI) in Reference 2. The attachment to this letter contains our response to the RAI.

Please contact Mike Ronski of my staff at (920) 388-8799, if you have any questions or require additional information.

A001

Sincerely,

for M. L. Marchi
Site Vice President-Kewaunee Plant

MAR/jmf

Attach.

cc - US NRC Senior Resident Inspector
US NRC Region III

ATTACHMENT

Letter from M. L. Marchi (WPSC)

To

Document Control Desk (NRC)

Dated

August 13, 1998

**Response to Request for Additional Information Related to
Proposed Technical Specificatiou Amendmeut 152**

This attachment provides responses to the NRC staff request for additional information (RAI) dated July 1, 1998.

NRC Question 1

As a result of the findings of the Maine Yankee Lessons Learned Task Force, the staff has taken a position that licensees must verify conformance to any conditions and limitations identified in the safety evaluation reports (SERs) that approved the analytical methodologies used in safety analyses supporting licensing applications. Accordingly, you are requested to confirm conformance to the conditions and limitations, if any, specified in the SERs for the methodologies used for the LOCA and non-LOCA transients analysis supporting the TS changes.

Response

Conformance to the Non-LOCA transient analysis methods conditions and limitations that are identified in SERs (Ref. 1,2) has been confirmed:

The VIPRE-01 application is limited to heat transfer modes up to critical heat flux.

Analyses have been performed to ensure that VIPRE-01 can predict the data base of DNB occurrence with at least a 95% probability at a 95% confidence level. Both the W-3 CHF correlation (Ref. 1) and the HTP CHF correlation (Ref. 2) have been analyzed and approved.

VIPRE-01 modeling assumptions, choice of two-phase flow models, correlations, and input values of plant-specific data are consistent with Ref. 1 approved values.

For transient calculations where the profile fit of the Levy subcooled boiling correlation is used, the Courant number is greater than 1.

WPS continues to abide by the EPRI Quality Assurance Program for the VIPRE-01 code.

The HTP CHF correlation with a safety limit of 1.14 is applied within the range of operating conditions for which the correlation and limit were approved (Ref. 2). Also, the fuel design characteristics for this safety analysis are within the correlation data base range (Ref. 2).

The W-3 correlation is used for Main Steam Line Break analysis since the operating conditions in this transient are outside the range analyzed for the HTP correlation.

The limiting FΔH for the most limiting dropped rod configuration for the Control Rod Drop Accident is determined using steady-state analysis; and the thermal margin at steady state is determined using subchannel analysis with VIPRE-01. This is acceptable since automatic rod control is administratively limited by constraints on power (less than 90%) and control rod bite (greater than 215 steps), and analyses of transients for which a trip does not occur (with concomitant power overshoots) are not necessary.

WPS does not analyze the Fuel Handling Accident or the Loss of Coolant Accident. The Small Break and Large Break Loss of Coolant Accidents have been analyzed by Westinghouse (Ref. 4 and 5). The Fuel Handling Accident has been analyzed by Framatome Cogema Fuels (Ref. 3).

Westinghouse has initiated a program to review the SERs for the codes and methodologies used for large break and small break LOCA analyses, and document the processes used to ensure compliance with any conditions or limitations. No non-conformances have been identified in the initial review of the methodologies used for the Kewaunee large and small break LOCA analyses. This program will be completed on a schedule which will support the industry's response to the forthcoming NRC Administrative Letter on this topic.

NRC Question 2

The non-LOCA analysis was performed for a full core design of SPC heavy fuel (Attachment 5, Page 2). However, it appears (Attachment 4, Page 2) that the proposed Cycle 23 core is a mixed core with both SPC heavy fuel and SPC standard fuel. Please clarify the types of fuel used in the core. If a mixed core is proposed as a transition core for operation, discuss the mixed core penalty on the DNBR calculations and the methods used for assessment. Present the results showing that the safety DNB limits are not violated for the applicable transients.

Response

Cycle 23 is a mixed core design. The types of fuel used in the core are presented in Table 1. SPC standard fuel with Bi-M grids and SPC standard fuel with HTP grids are both in the core. All of the SPC heavy fuel in the core contains HTP grids.

WPS has performed mixed core thermal hydraulic analyses using the VIPRE-01 code. Results of these mixed core analyses indicate that a standard fuel assembly with Bi-M grids that has four face-adjacent fuel assemblies with HTP grids experiences a 3.3% reduction in DNBR. This is expected since a Bi-M grid has a slightly higher flow resistance than an HTP grid. The fuel rods in SPC standard and SPC heavy fuel have the same outer diameter and therefore have matching fuel rod related flow resistances. The other components of SPC standard and SPC heavy fuel assemblies also have matching flow resistances.

WPS conservatively applies a mixed core DNBR penalty of 3.3% in the safety analyses to the approved W-3 correlation limit for standard fuel with Bi-M grids. As a result, the Bi-M grid standard fuel assemblies continue to have full power peaking factor limits of 1.55 FΔH and 2.28 FQ. Safety analysis results for the SPC standard fuel assemblies with Bi-M grids are presented in Table 2.

Standard or heavy fuel assemblies with HTP grids, on the other hand, experience a DNBR improvement in a mixed core design with Bi-M grid standard fuel assemblies due to the reduced flow resistance of the HTP grids. No DNBR penalty is therefore needed for SPC standard or heavy fuel with HTP grids. However, the standard fuel assemblies with HTP grids will conservatively use the same full power peaking factor limits of 1.55 FΔH and 2.28 FQ that are used for the SPC standard fuel with Bi-M grids. Safety analyses for the SPC heavy fuel assemblies with HTP grids are performed assuming a full core of fuel assemblies of that design. DNBR results from the safety analyses of the SPC heavy fuel assemblies with HTP grids are presented in Attachment 5 of Ref. 7.

Periodic core surveillance ensures that the fuel assemblies satisfy their respective power peaking factor limits.

Safety analysis results for fuel designs used in Cycle 23 show that DNBR limits with consideration for mixed core thermal hydraulic effects are satisfied for all applicable transients.

NRC Question 3

The discussions of the LOCA analyses does not identify the specific breaks that were analyzed and the limiting breaks for both small break and large break LOCA cases. Provide this information and explain your rationale for concluding that the limiting cases were identified and analyzed.

Response

A range of break sizes was analyzed for small break LOCA. The limiting break (i.e., the highest peak clad temperature) was found to be a 3-inch diameter cold leg break (Ref. 5).

In Ref. 7, Sections 6-1 and 7-4.2, break spectrum sensitivity study results for LBLOCA are presented. From these sensitivity studies, using the WCOBRA/TRAC upper plenum injection models, it is concluded that the limiting break size and location is the 0.6 double-ended cold leg (DECLG) break for superbounded and 0.4 DECLG break for Appendix K LOCA analyses. The rationale is that these break sizes and locations yield the highest PCTs.

NRC Question 4

The application does not include information for the steam generator tube rupture (SGTR) event. Explain why the SGTR event was not considered in the application.

Response

The Kewaunee Steam Generator Tube Rupture (SGTR) analysis has been reviewed and found to not be sensitive to the fuel design or peaking factor limit changes that are proposed in Ref. 7. In addition, the SGTR analysis source term adequately bounds the Kewaunee core designs that are anticipated with Ref. 7 approved. Current safety analysis for SGTR is also bounding with respect to the change in reactor coolant system flow limit that is proposed in Ref. 7. The reactor coolant system flow limit reduction (proposed in Ref. 7) is a change that improves the safety analysis results of a SGTR accident, since it would tend to reduce the primary to secondary break flow during the SGTR transient, thus reducing the overall accident consequences.

NRC Question 5

For the locked rotor event, the application states that the percentage of fuel rods experiencing DNB depends on the reload calculation; however, no number was provided (Attachment 5, Page 5). Please provide information to show that the acceptance criteria in SRP 15.3.3 for the locked rotor event are met. Specifically, acceptance criterion II.2 requires that "Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability," and acceptance criterion II.3 requires that "Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines."

Response

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 which is 40%. This acceptance criterion and the associated radiological consequences assessment consistent with Kewaunee's licensing basis have been previously reviewed and accepted by the NRC in References 1, 8, 9, and 10. Therefore, WPSC has previously addressed the criteria II.2 and II.3 of SRP 15.3.3.

References

1. WPSRSEM-NP-A, Revision 2, October 1988, Reload Safety Evaluation Methods for Application to Kewaunee

2. Letter from R. J. Laufer to M. L. Marchi, Dated 12/30/97, "Safety Evaluation for the High Thermal Performance Departure from Nucleate Boiling (DNB) Correlation and the Associated 1.14 Minimum DNB Ratio Safety Limit for Kewaunee Fuel"

3. Framatome Cogema Fuels Document 51-1267175-00, "Kewaunee New and Spent Fuel Storage Rack Licensing Report," May 1998

4. Letter from Stephen Swigart (Westinghouse) to D. J. Wanner (WPS) transmitting "Kewaunee Final Large Break LOCA USAR Updates for SPC 14 x 14 Heavy Fuel," June 18, 1998

5. WCAP-14103, "Westinghouse Small Break LOCA Kewaunee NOTRUMP Analysis," June 1994

6. WCAP-10924 P-A, Rev. 2, "Westinghouse Large Break LOCA Best Estimate Methodology," December 1988

7. Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC), Dated 4/15/98, transmitting Proposed Amendment 152 to KNPP Technical Specifications

8. Letter from A. Schwencer (NRC) to E. W. James (WPSC) dated May 25, 1978, Safety Evaluation for License Amendment 21.

9. Letter from A. Schwencer (NRC) to E. W. James (WPSC) dated October 22, 1979, Safety Evaluation for Qualification of Reactor Physics Methods.

10. Letter from J. Giitter (NRC) to D. C. Hintz (WPSC) dated April 11, 1988, Safety Evaluation for Reload Safety Evaluation Methods.

TABLE 1

FUEL LOADED INTO CYCLE 23

REGION	NO. OF ASSEMBLIES	NO. OF DUTY CYCLES	CYCLE 22 STATUS	FUEL VENDOR	INITIAL U235 ENRICHMENT	FUEL ROD DESIGN	GRID DESIGN
20	1	2	Spent Fuel Pool	SPC	3.4	Standard	Bi-M
22	8	3	Reactor	SPC	3.7	Standard	Bi-M
23	4	2	Reactor	SPC	3.8	Standard	Bi-M
23	12	2	Reactor	SPC	4.1	Standard	Bi-M
23	8	2	Reactor	SPC	4.1	Heavy	HTP
24	24	1	Reactor	SPC	4.1	Standard	HTP
24	12	1	Reactor	SPC	4.5	Standard	HTP
24	8	1	Reactor	SPC	4.5	Heavy	HTP
25	20	0	Fuel Vendor	SPC	4.1	Heavy	HTP
25	24	0	Fuel Vendor	SPC	4.5	Heavy	HTP
TOTAL	121						

Bi-M denotes the SPC Bi-Metallic grid design.

HTP denotes the SPC High Thermal Performance grid design.

TABLE 2
NON-LOCA SAFETY ANALYSIS

CONDITION II AND CONDITION III EVENTS

		Calculated Value/Acceptance Criteria		
USAR Section	Transient	MDNBR	Reactor Coolant System Pressure (psia)	Main Steam System Pressure (psia)
14.1.1	Uncontrolled Rod Withdrawal from Subcritical	3.85/1.34	2341/2750	1104/1210
14.1.2	Uncontrolled Rod Withdrawal at Power			
	Fast Rate Full Power	1.93/1.34	2269/2750	938/1210
	Slow Rate Full Power	1.64/1.34	2321/2750	917/1210
	Fast Rate Intermediate Power	2.37/1.34	2331/2750	954/1210
	Slow Rate Intermediate Power	1.36/1.34	2351/2750	1182/1210
14.1.3	Control Rod Misalignment $F\Delta H$=1.85	1.49/1.34	2220/2750	750/1210
14.1.4	Chemical and Volume Control System Malfunction	1.62/1.34	2326/2750	923/1210
14.1.5	Startup of Inactive Loop	10.0/1.34	2320/2750	1149/1210
14.1.6	Feedwater System Malfunction			
	BOC Manual Control	2.08/1.34	2220/2750	750/1210
	EOC Auto Control	2.03/1.34	2220/2750	750/1210
14.1.7	Excessive Load Increase			
	BOC Manual Control	2.08/1.34	2220/2750	750/1210
	BOC Auto Control	1.78/1.34	2230/2750	750/1210
	EOC Manual Control	1.85/1.34	2220/2750	750/1210
	EOC Auto Control	1.81/1.34	2221/2750	750/1210
14.1.8	Loss of Flow			
	2/2 Pump Trip	1.78/1.34	2311/2750	907/1210
	Underfrequency Trip	1.77/1.34	2312/2750	907/1210
14.1.9	Loss of Load			
	BOC Manual Control	2.08/1.34	2501/2750	1182/1210
	BOC Auto Control	2.08/1.34	2469/2750	1182/1210
	EOC Manual Control	2.08/1.34	2479/2750	1171/1210
	EOC Auto Control	2.08/1.34	2383/2750	1198/1210
14.1.10	Loss of Feedwater	2.08/1.34	2233/2750	1093/1210

TABLE 2
NON-LOCA SAFETY ANALYSIS

CONDITION IV EVENTS

		Calculated Value/ Acceptance Criteria	
USAR Section	Transient	MDNBR	Containment Pressure (psia)
14.2.5	Main Steam Line Break (Core Response, FΔH = 5.25) Upstream Flow Restrictor	1.52/1.50	---
	Main Steam Line Break (Containment Response) SLB14MYY0	---	60.7/60.7

		Calculated Value/Acceptance Criteria				
USAR Section	Transient	Max Clad Temp. (°F)	Max Fuel Centerline Temp. (°F)	Max Energy Deposition (cal/grm)	RCS Pressure (psia)	MSS Pressure (psia)
14.2.6	Control Rod Ejection					
	BOC Full Power	2044/2700	4658/4700	154/200	2295/2750	863/1210
	BOC Zero Power	2551/2700	4040/4700	148/200	2396/2750	1076/1210
	EOC Full Power	2022/2700	4641/4700	153/200	2308/2750	864/1210
	EOC Zero Power	2680/2700	4147/4700	154/200	2333/2750	1051/1210

		Calculated Value/Acceptance Criteria			
USAR Section	Transient	% Fuel Rods < DNB Limit	Max Clad Temp. (°F)	RCS Pressure (psia)	MSS Pressure (psia)
14.1.8	Locked Rotor	*0.0/40	1499/2700	2385/2750	1044/1210

* Cycle 23 standard fuel rods with FΔH ≥ 1.42