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SUBJECT: Application to amend License NPF-21,requests changes in Tech Specs to allow final feedwater reduction of 65% power.

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September 14, 1988
G02-88-198

Docket No. 50-397

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Gentlemen:

Subject: NUCLEAR PLANT NO. 2
OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATIONS
FINAL FEEDWATER TEMPERATURE REDUCTION (FFTR)

Reference: Letter, G02-87-286, GC Sorensen to NRC,
same subject, dated December 15, 1987

In the reference letter, the Supply System requested an amendment to the WNP-2 Technical Specifications (Tech. Specs.) to allow the operation of WNP-2 with final feedwater temperature reduction and subsequent thermal coastdown to 65% power for the purpose of extending the normal fuel cycle. In telephone conversations with NRC staff, a number of questions were asked of the Supply System regarding other plant's experience with final feedwater temperature reduction and with regard to the specific mechanical design of the WNP-2 feedwater nozzles and spargers. The Staff requested that the Supply System summarize the NRC questions and Supply System responses in letter form. Those questions and answers are submitted herewith as Attachment 1.

In subsequent reviews of the transient analyses submitted with the reference, the NRC staff determined that the submitted analyses justified the requested Tech. Spec. Critical Power Ratio (CPR) values for Cycle 3. However, they felt that additional analyses would be appropriate in order to justify approval of the requested CPR values for Cycle 4 and beyond (i.e., generic approval). The Supply System agreed to direct and perform additional analysis of final feedwater temperature reduction and thermal coastdown for Cycle 4, it being the apparent consensus that, if Cycle 4 results proved to be bounded by Cycle 3 results, the basis for generic approval would be in place.

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Attached to this letter (Attachment 2) are the results of the final feedwater and thermal coastdown analysis for Cycle 4 ((XN-NF-87-92), Supplement 1, dated May 1988). The results of the appended analysis demonstrate that the requested CPR values based on Cycle 3 analysis bound the Cycle 4 analysis (see page 12, XN-NF-87-92, Supplement 1). The fact that Cycle 3 CPR values bound Cycle 4 CPR values for final feedwater temperature reduction was expected due to the changes in physical phenomena taking place in the WNP-2 core because the Cycle 4 reload core is moving toward and is much more representative of the equilibrium core than is the Cycle 3 core. The submitted CPR values have been demonstrated, by this and previously submitted analyses, to be conservative and bounding for final feedwater temperature reduction operation.

One of the attachments to the reference letter was a report (NEDC-31107) which the Supply System submitted to support the mechanical ability of the feedwater system to operate in the proposed manner. In reviewing this report, the NRC staff has raised questions relating to Table 2-1, p. 2-6, report, titled "Core-Wide Transient Analysis Results at ICF and/or FFWTR." It is the Supply System's belief that the results of this table are not relevant to the matter at hand. They represent analyses performed by a different vendor, with different methods, on a different core. They were not cited by the Supply System in its application and only happen to be a part of the application because they were physically in the same document referenced by the Supply System to supply the mechanical characteristics of the feedwater system for the proposed operational mode.

However, in the interests of completeness, the Supply System has reviewed the staff questions and has the following comments. In NEDC-31107, the Load Rejection with No By Pass (LRNBP in General Electric terminology; Load Rejection Without Bypass (LRWB) in Advanced Nuclear Fuels terminology) and Feedwater Controller Failure (FWCF) transients are analyzed at 100% power, 106% flow for normal and FFTR conditions. This analysis (of the WNP-2 initial core) indicates essentially no change in delta CPR when FFTR is utilized. In XN-NF-87-92, the other attachment to the reference letter which was submitted by the Supply System as the reference transient analysis for the proposed operation, the effect of FFTR is shown to increase the delta CPR for the LRWB by up to .02 and to decrease the delta CPR for the FWCF event as much as .01.

The Supply System believes that the above phenomena are consistent, explainable and serve to demonstrate that the analysis submitted by the Supply System is conservative relative to the actual event. The LRWB is recognized as the limiting transient in the Supply System submittal. The analysis reported in NEDC-31107 for the LRWB was performed by taking an end-of-cycle Haling power distribution, reducing final feedwater temperature and then calculating a delta CPR using the resultant Haling distribution. Reducing the final feedwater temperature tends to cause a very small effect on delta CPR when analyzed in this manner. That is the case as shown in NEDC-31107. The analysis performed in XN-NF-87-92 for the LRWB was performed by taking an end-of-cycle Haling power distribution, reducing final feedwater temperature, doing an additional burnup step and then calculating delta CPR. The effect of the burnup step is to move power further up the core, hence reducing the effect of the terminating scram, and thus significantly increasing delta CPR. That is the case as shown in XN-NF-87-92 for the LRWB. The FWCF transient was analyzed at different, and, we believe, more conservative conditions in XN-NF-87-92 relative to the analysis reported in NEDC-31107. Therefore, the results are not directly comparable. In any case, as stated above, the LRWB is demonstrated to be the limiting transient for WNP-2 in FFTR operation. The analysis provided in XN-NF-87-92, Supplement 1 is clearly conservative and bounding.

Very truly yours,

GC Sorensen

G. C. Sorensen, Manager
Regulatory Programs

HLA/bk
Attachments (As stated)

cc: JB Martin - NRC RV
NS Reynolds - BCP&R
RB Samworth - NRC
DL Williams - BPA/399
NRC Site Inspector - 901A

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ATTACHMENT 1

Questions and Answers regarding other plant's experience with Final Feedwater Temperature Reduction.

QUESTION 1

For Paragraph 3.2.1 of NEDC-31107 (attached to reference letter), the NRC wanted assurance that the basis approved in the FSAR has not been revised via new or different allowance in the ASME Section III, Subsection NG.

RESPONSE

There was no change in the design allowable or basis with the final feedwater temperature reduction as originally defined in the WNP-2 FSAR and the ASME Section III, Subsection NG.

QUESTION 2

In Section 4 of NEDC-31107, last paragraph, the NRC stated that this section was too vague. They wanted confirmation that the maximum flow is 106% and definition as to the vibration level at this condition. Also, they wanted clarification on the relationship of the vibration requirement between Tokai and WNP-2.

RESPONSE

Vibration data from Tokai 2 (BWR-5/251 prototype plant), which was tested up to 100% core flow, was evaluated and used to identify the components with the highest flow-induced vibration stresses. Data from WNP-2 testing up to 106% core flow were reviewed for these components. The maximum alternating stress intensity, determined from the WNP-2 tests for core flow up to 106%, is 92% of the acceptance criterion of 10,000 psi.

The maximum stress intensity was determined by absolute summation of the individual percent criteria for each vibration mode. This analysis method is conservative in that the criteria are based on the assumption of vibration at a constant sustained maximum amplitude for each mode, whereas actual vibration amplitudes are generally random and seldom reach the maximum recorded values. An additional conservative factor is the use of 10,000 psi as the maximum allowable peak stress amplitude for sustained vibration stress for stainless steel. The 10,000 psi is more conservative than the current ASME Section III allowable stress of 13,600 psi for 10^{11} cycles.

QUESTION 3

In Section 5.3 of NEDC-31107, does the nozzle and sparger design meet the provisions of NUREG 0619? Also, the NRC staff members said the statement regarding stainless steel being "less susceptible to high cycle fatigue than low alloy steel" is not correct for the operating condition and design of the feedwater sparger. They stated that stainless steel would be more susceptible due to the thermal gradient properties and the thermal stress would be higher due to the higher thermal expansion coefficient.

RESPONSE

The Hanford 2 (WNP-2) welded sparger design meets the provisions of NUREG 0619, Section 4.1, Item (3).

The general statement that stainless steel is "less susceptible to high cycle fatigue than low alloy steel" comes from inspection of the two respective fatigue curves for these materials. For a given stress, stainless steel can tolerate more cycles than low alloy steel before fatigue failure occurs. (Fatigue curves of ASME Code, Section III, Appendix I). However, it is recognized that stainless steel, as compared to an identical carbon steel part, will yield a higher overall thermal stress state for the same thermal boundary conditions and thermal restraints. As a result, a parametric study comparing the fatigue response of carbon and stainless steel sparger designs was completed to demonstrate that stainless steel has a greater fatigue integrity in this application. The parametric study is included in the design record file supporting development of NEDC 31107.

QUESTION 4

The NRC wanted identification of which other plants have implemented feedwater temperature reduction and have the welded sparger designs.

RESPONSE

Plants with welded sparger design which have implemented FFWTR or FWHOS (Feedwater Heater Out of Service) include Brunswick 1 and Hatch 2.