TU ELECTRIC

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William J. Cahill, Jr.
Group Vice President

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 2
DOCKET NO. 50-446
REQUEST FOR ADDITIONAL INFORMATION ON CPSES FINAL SAFETY ANALYSIS REPORT (FSAR) CHAPTERS 4 AND 15, AMENDMENTS 83 AND 84

REF: (1) NRC Letter from Mr. Brian E. Holian to Mr. William J. Cahill Jr., dated July 20, 1992
(2) Comanche Peak Steam Electric Station, Unit 1 - Amendment No. 10 to Facility Operating License No. NPF-87, dated June 8, 1992

Gentlemen:

TU Electric's responses to the nine questions in reference 1 are attached. Two additional questions, which were asked by Mr. Tai Huang at the meeting in Rockville, Maryland on June 4, 1992, are also included as questions 10 and 11. Should clarification or additional information be required to enable the NRC Staff to complete its review, please call David Bize at (214) 812-8879.

Sincerely,

William J. Cahill, Jr.

By: D.R. Woodlan
D. R. Woodlan
Docket Licensing Manager

DNB/dnb
Attachment
Enclosure

c - Mr. J. L. Milhoan, Region IV
Resident Inspectors, CPSES (2)
Mr. T. A. Bergman, NRR
Mr. B. E. Holian, NRR

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ADDITIONAL INFORMATION REGARDING

CHAPTERS 4 AND 15 OF FINAL SAFETY ANALYSIS REPORT

COMANCHE PEAK STEAM ELECTRIC STATION UNIT 2

1. Question

Section 4.2.2.3 of Chapter 4 of the FSAR, page 4.2-21, mentions the desirability of a negative moderator coefficient when greater than 75 percent of full power. However, Figure 15.0.6 shows a positive value up to 100 percent of full power. Explain this discrepancy.

Response

Section 4.2.2.3 of the FSAR will be revised in Amendment 86, consistent with Amendment 6 of Technical Specifications, to reflect that the moderator temperature coefficient may be positive at power levels below 100% Rated Thermal Power.

2. Question

Use of hafnium as the absorber material in the control rods is mentioned throughout Chapter 4. NRC Information Notice No. 89-31, "Swelling and Cracking of Hafnium Control Rods," alerted PWR licensees of swelling and cracking of hafnium control rods at several PWRs. Did you consider this information in your application of hafnium as a control rod material for Unit 2?

Response

Ag-In-Cd control rods are used in CPSES as shown in Table 4.1-1B and stated on page 4.2-12. Section 4.2.2.3.1 of the FSAR will be revised in Amendment 86 to reflect that Ag-In-Cd alloy is the primary design and hafnium, the alternate design.

3. Question

Section 4.3.2.2.8 of Chapter 4 of the FSAR specifies that tests performed at the beginning of each reload cycle are limited to verification of steady state power distributions. Explain why control rod worth measurements and moderator temperature coefficient surveillance are not also performed at this time.

Response

Section 4.3.2.2.8 was revised in Amendment 85 to acknowledge the tests performed at the beginning of each reload cycle.

4. Question

Section 4.3.2.6 of Chapter 4 of the FSAR refers to the use of the LEOPARD and PDQ computer codes for fuel storage criticality calculations. NRC Information Notice 92-21, "Spent Fuel Pool Reactivity Calculations," indicates inaccuracies discovered in the use of these codes to predict the criticality in fuel storage racks. Did you consider this information on potential computer code inaccuracies in relation to your Unit 2 fuel storage analyses?

Response

TU Electric has obtained confirmation from Westinghouse that the enrichment limits set with LEOPARD and PDQ contain substantial conservatism and that higher enrichment limits can be justified using more sophisticated analysis. IN 92-21 was mainly concerned with the use of Boraflex, which is not employed at CPSES.

5. Question

The first footnote to Table 4.3-[3]B of Chapter 4 of the FSAR for Unit 2 refers to a value which includes a 0.1 percent delta-rho uncertainty. What value is being referred to?

Response

The footnote applies to the Doppler defect. The omission will be corrected in FSAR Amendment 86.

6. Question

Table 4.3-4 of Chapter 4 of the FSAR which is supposed to summarize the comparisons of criticality calculations with 101 critical experiments is missing.

Response

Table 4.3-4 is still listed as an effective page in the current FSAR. A copy of the table is enclosed.

7. Question

Explain why the control rod drop time has decreased to 2.4 seconds for Unit 2 compared to 3.3 seconds for Unit 1.

Response

Current Comanche Peak Units 1 and 2 Combined Technical Specifications (Proof and Review copy for Unit 2) require the measured control rod drop time to be less than or equal to 2.4 seconds.

In some analyses for Unit 1, Westinghouse used a control rod drop time of 3.33 seconds for the FSAR non-LOCA analyses assuming that the control rods at CPSES would be B_4C control rods. The B_4C control rods were never used and were replaced by Ag-In-Cd control rods which have a faster drop time than the B_4C control rods. Ag-In-Cd control rods are also installed for Unit 2. Thus, any control rod drop time greater than or equal to 2.4 seconds is acceptable for use in the Unit 2 analyses.

8. Question

The analysis for the uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition [assumes two reactor coolant pumps to be in operation. Technical] Specifications allow fewer than two pumps to be in operation during shutdown. The analysis should be performed from flow conditions corresponding to the minimum number of allowable operating pumps.

Response

The FSAR presents an analysis for the uncontrolled rod cluster control assembly bank withdrawal event in Mode 2. An occurrence in Modes 3, 4 or 5 with two or more reactor coolant pumps in operation would be bounded by the analysis in Mode 2. This is based upon the FSAR analysis assumption that reactor trip does not occur until the power-range (low setting) high neutron flux setpoint is reached and that two banks are withdrawn sequentially at maximum speed (72 step/min). These conservative assumptions result in the core returning to critical and generating some power prior to trip. Therefore, the primary system flow rate becomes an important consideration as a factor in DNB evaluation. (Note that in Mode 3, the Technical Specification 3.4.1.2 requires that at least two reactor coolant pumps to be in operation whenever the reactor trip breakers are closed.)

In Modes 3, 4 and 5, the source range high neutron flux trip will be available (Technical Specification 3.3.1, Table 3.3-1) to terminate the event by tripping any withdrawn and withdrawing rods before any significant power level could be attained. Also, the reactivity insertion rate would be slower because a failure in the rod control system could cause, at most, the withdrawal of only one bank and its withdrawal rate would be slower than the maximum rod speed which is possible when in automatic rod control. Under these conditions, DNB (and fuel failure) would not be credible.

9. Question

Recent nonconservatisms were identified at Comanche Peak related to the input assumptions and boundary conditions (inverse count rate ratio data and flux-multiplication setpoint) in the analyses of the licensing basis boron dilution event. Based on this, justify the automatic actions to terminate the dilution and start boration which were assumed in the boron dilution analyses for Unit 2.

Response

The CPSES Unit 2 licensing analysis for the Boron Dilution event has been completed and is consistent with the analysis supporting a License Amendment (reference 2) which was granted for CPSES Unit 1. The License Amendment temporarily removes the operability requirement for the Boron Dilution Mitigation System (BDMS) from the Technical Specifications. CPSES Unit 1 BDMS operability will not be required until six months following the second refueling outage for CPSES Unit 1. Similarly, a six month evaluation period of the BDMS following initial criticality has been requested for the draft Unit 2 Technical Specifications. The basis for the License Amendment was a revised Boron Dilution analysis and additional administrative controls.

The revised Boron Dilution analysis for Unit 1 and Unit 2 does not credit automatic actions of the BDMS. The licensing analyses for Modes 3, 4 and 5 demonstrate that there are at least 15 minutes from the initiation of a boron dilution before shutdown margin is lost. These analyses provide reasonable confidence that the reactor operators have sufficient time during performance of their routine duties to identify and mitigate a boron dilution event.

In addition to the revised analyses, the following compensatory actions are specified for the duration of the temporary Technical Specification revision:

1) Within 4 hours of entry into MODES 3, 4, or 5 from MODES 1, 2, or 6, (and once per every 14 days thereafter while in MODES 3, 4, or 5), TU Electric will verify (unless startup is in progress) that either valve CS-8455 or valves CS-8560, FCV-111B, CS-8439, CS-8441, and CS-8453 are closed and secured in position; or

2) Following entry into MODES 3, 4, or 5 from MODES 1, 2, or 6, each crew of the Control Room Staff will receive a briefing to discuss the type of reactivity changes that could occur during a dilution event; the indication of a dilution event; and the actions required to stop dilution, commence immediate boration and establish the required shutdown margin. For extended shutdowns, this briefing will be repeated for each crew prior to resumption of control room duties following an off duty period which exceeds 7 days. During time periods when this option is used the source range will be monitored for indication of unexplained increasing counts and inadvertent boron dilution every fifteen (15) minutes. In addition, within 4 hours of entering MODE 5, TU Electric will ensure that only one Reactor Makeup Water Pump (dilution source) is aligned to the supply header.

Even though credit is not taken for the BDMS, its use during plant operation provides additional assurance that an inadvertent dilution event will be detected and mitigated prior to a return to critical. In addition, other alarms and indications (as provided in Section 15.4.6.1 of the CPSES FSAR) are available to the operator which allow for the detection of an inadvertent boron dilution.

In view of these alarms and indications, together with the procedures, training, and activities previously mentioned, reasonable assurance has been provided to minimize the likelihood of an inadvertent boron dilution event during the time interval for the temporary TS revisions. Should such an event occur, these actions provide reasonable assurance of timely detection and mitigation.

10. Question

When comparing Figures 4.3-22A and 4.3-22B, why is the distribution of peak linear power, kw/ft, shifted about the point of delta I = 0?

Response

In Figure 4.3-22A, Unit 1, some of the data for the end-of-life load follow cases were inadvertently omitted. If those points are included, then Figure 4.3-22A will look similar to Figure 4.3-22B which is correct for Unit 2.

11. Question

BOL and EOL Doppler temperature coefficient curves, Figures 4.3-27A and 4.3-27B, are different. What differences between OFA and standard fuel design or operating characteristics cause the differences in the shape and orientation of the curves?

Response

The Unit 1 curve, Figure 4.3-27A, was generated with an earlier methodology than the Unit 2 curve. If the curves for both Units are calculated with a spatial weighting factor applied, they exhibit similar characteristics. Incidentally, Figure 4.3-27B will be updated to reflect CPSES Unit 2 specific results, with 3-D spatial weighting to be included.

TABLE 4.3-4

BENCHMARK CRITICAL EXPERIMENTS

Description of Experiments*	Number of Experiments	LEOPARD k_{eff} Using Experimental Bucklings
UO_2		
Al clad	14	1.0012
SS clad	19	0.9963
Borated H_2O	7	0.9989
Subtotal	40	0.9985
U-Metal		
Al clad	41	0.9995
Unclad	20	0.9990
Subtotal	61	0.9993
Total	101	0.9990

* Reported in Reference [12].