



Omaha Public Power District

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June 26, 1980

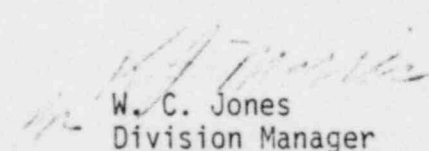
Director of Nuclear Reactor Regulation
ATTN: Mr. Robert A. Clark, Chief
Operating Reactors Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: Docket No. 50-285

Gentlemen:

The Omaha Public Power District received a telecopy from the Commission containing questions in regard to the report entitled "Fort Calhoun Cycle 6 Small Break ECCS Performance Evaluation at 1500 MWt", which was submitted to the Commission on May 21, 1980, in support of an Application for Amendment of Operating License ("Stretch Application") filed on July 17, 1979. Attached hereto are responses to the telecopy questions.

Sincerely,


W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH:jmm

Attachment

cc: Mr. Philip C. Wagner
Project Manager
U. S. Nuclear Regulatory Commission
Mail Stop 228
Washington, D. C. 20555

LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

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Responses to Questions on
Small Break LOCA

Question 1

Provide Reference 8.

Response

Reference 8 of the submittal is "Letter from T. E. Short to George E. Lear dated December 12, 1976".

Question 2

Provide statement and appropriate documentation that this evaluation is in conformance with Appendix K of 10CFR50.

Response

Reference 1 provides the results of the Fort Calhoun small break LOCA ECCS performance analysis at 1530 MWt. The computer programs and calculative methods employed in the analysis were approved by the NRC and are identified in Reference 1. The references for these computer programs and calculative methods listed in Reference 1 are presented again in this response as References 2 through 5 and 10. The NRC's safety evaluation report, documenting approval of these codes and calculative methods, is presented in References 6 through 9. The NRC's safety evaluation reports provide the appropriate documentation that this evaluation is in conformance with Appendix K of 10CFR50.

Question 3

Provide justification that the break spectrum between 0.5 and 0.1 ft is not limiting. That is, further discussion that verifies the parameters of importance to the breaks at either end of this spectrum are at the extremes of their influence and that no new phenomena become significant for the intervening breaks, if required.

Response

The major reasons for the non-limiting nature or low peak clad temperatures for breaks between 0.5 and 0.1 ft² are as follows. As the break size increases above 0.1 ft², the depressurization rates increase. The higher depressurization rates result in increased steam production during the blowdown due to increased flashing of the coolant which produces greater two phase level swell. Although the fluid inventory loss increases as the break size increases, the dominant phenomena is the increased level swell as a consequence of the higher depressurization rates. Thus, these higher depressurization rates minimize core uncover for breaks greater than 0.1 ft² and smaller than 0.5 ft². Also, for breaks in this range, the depressurization rates are also sufficient to actuate the safety injection tanks which quickly terminate the clad temperature transient and minimize the duration of the core uncover period. The high pressure safety injection pumps therefore do not play a significant role in affecting core uncover for these breaks.

Response (Continued)

As the break size approaches 0.5 ft², the inventory loss becomes the dominating phenomenon. As a consequence, as the break size increases, the core two phase level transients are characterized by increasing core uncover. This results in an increasing peak clad temperature characteristic as the break size approaches 0.5 ft².

Break sizes 0.1 ft² and smaller are characterized by slower depressurization rates which delay or prevent safety injection tanks from actuating. The core two phase level transient and hence peak clad temperature for these break sizes are therefore controlled by the high pressure safety injection pumps.

Reference 10 supports this discussion that breaks between 0.5 and 0.1 ft² are non-limiting based on a spectral analysis of a range of breaks from 1.0 to 0.03 ft² in area which demonstrated the limiting peak clad temperature to occur at the 0.1 ft² break.

In summary, as a consequence of the higher depressurization rates, shorter blowdown times and increased core two phase level swell, breaks less than 0.5 ft² and greater than 0.1 ft² are not limiting for the Fort Calhoun plant.

Question 4

Provide justification, preferably experimental, for the "corrected low pressure SI pump flow" which has been assumed in the 0.5 ft break evaluation. Assure that the justification applies to the range of operational conditions under the hypothesized LOCA. Provide evaluation of the uncertainties in pump performance characteristics.

Response

The high and low pressure safety injection performance curves, used in the small break LOCA analysis at 1530 MWt presented in Reference 1, were based on experimentally measured head and flow parameters. The safety injection flow credited in the Reference 1 analysis includes a 1% uncertainty in head and a 3% uncertainty in flow. The safety injection performance curves used in this analysis are shown in Figures 1 and 2.

Question 5

Provide documentation that the requirements of NUREG-0635 have been satisfied.

Response

The District is implementing the recommendations of NUREG-0635. Detailed below are the major recommendation areas of NUREG-0635 and the District's action.

FIGURE 1
FAILURE CONDITION - LOSS OF AN EMERGENCY GENERATOR

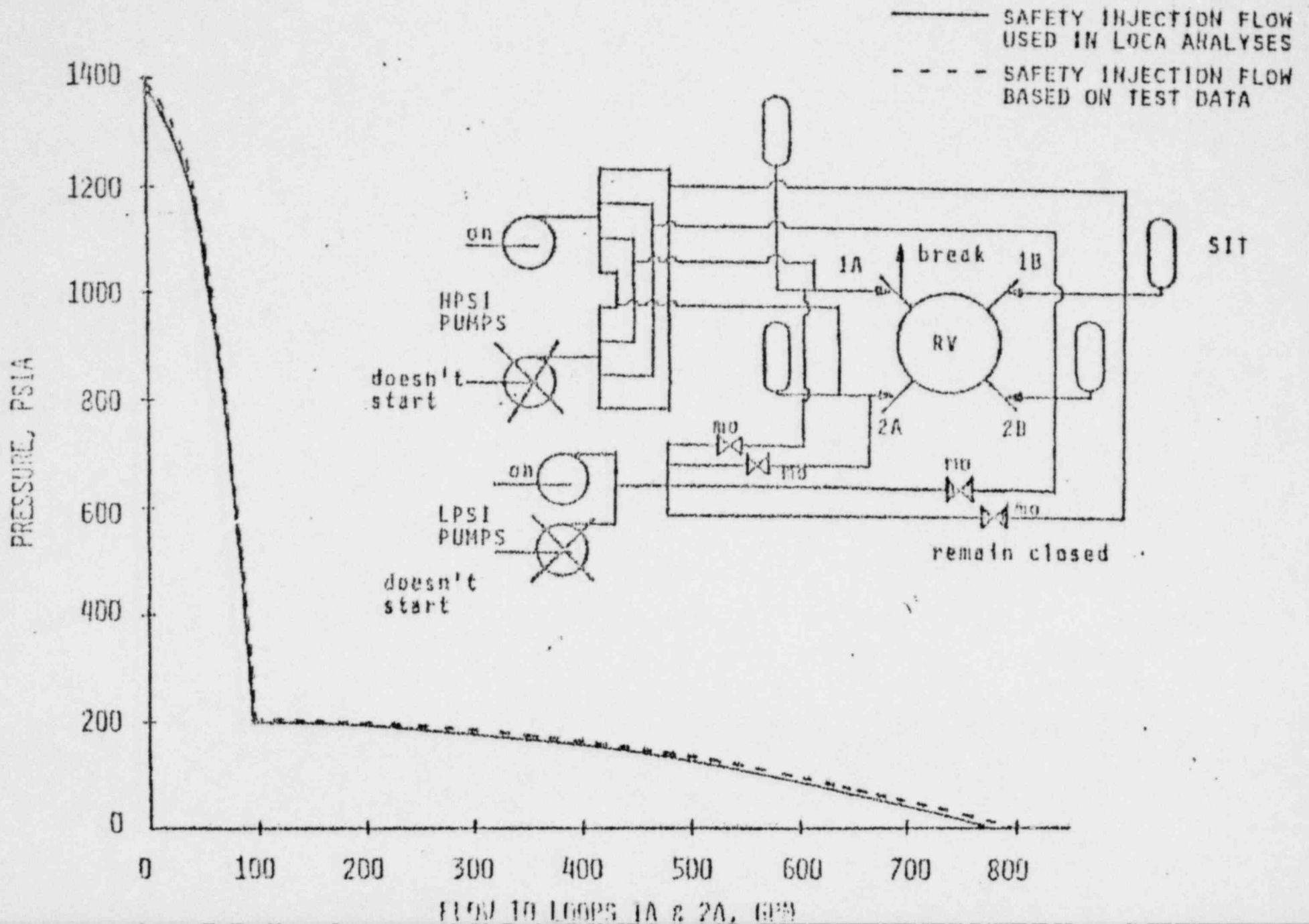
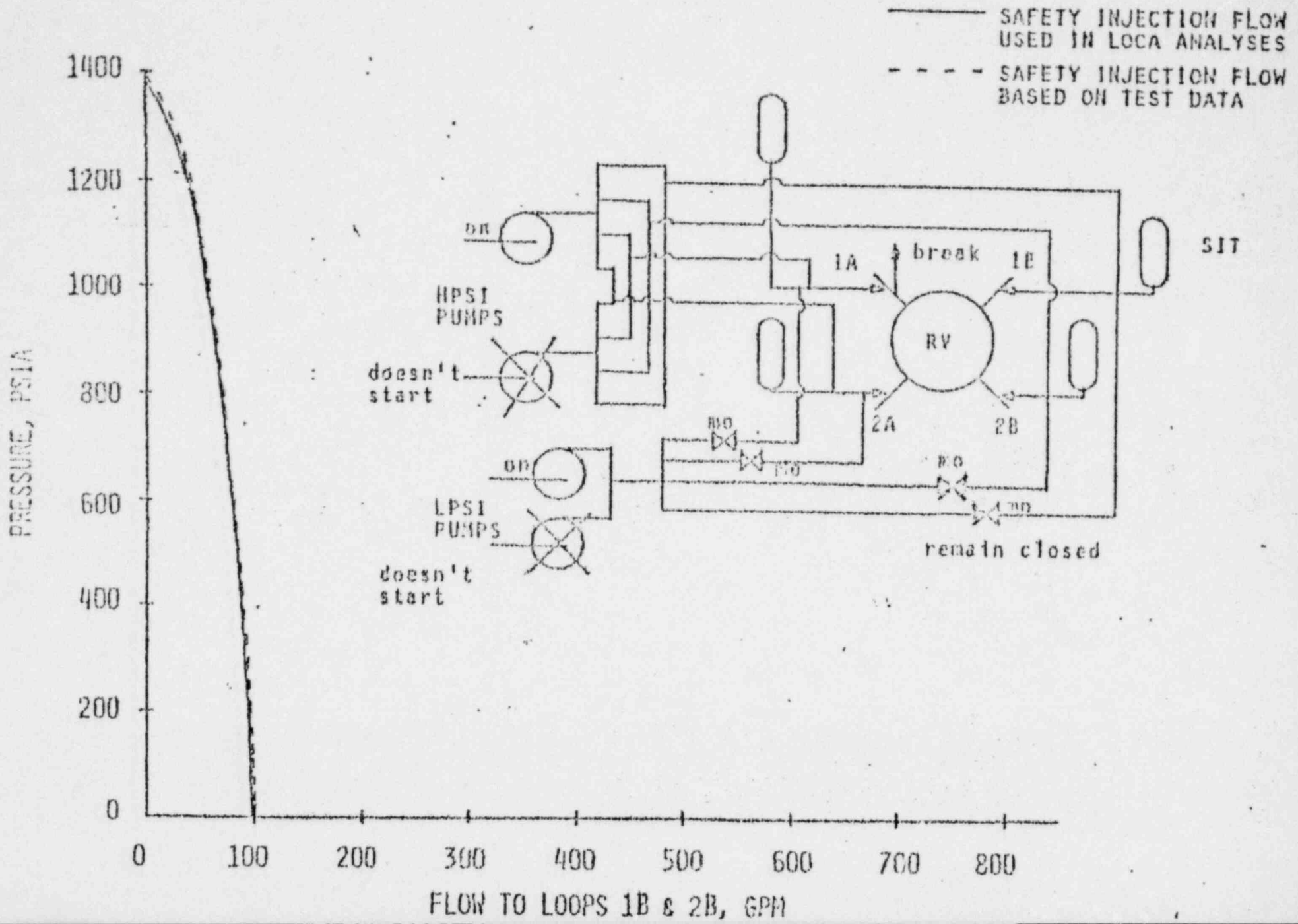


FIGURE 2
FAILURE CONDITION - LOSS OF AN EMERGENCY GENERATOR



Response (Continued)

Auxiliary Feedwater System

A letter from Darrell G. Eisenhut to W. C. Jones, dated October 22, 1979, specified the auxiliary feedwater system requirements for Fort Calhoun Station Unit No. 1. The District has addressed the short term requirements and is in the process of addressing the long term requirements.

Small Break LOCA Methods

The submitted small break LOCA analysis uses the most recently approved small break LOCA model for the core when the reactor coolant pumps are assumed to be tripped. This assumption is consistent with current operating procedures. We believe this model to be the best available licensing model.

The District is participating in the CE Owners Group for confirmation of the small break LOCA analysis methods. The District will submit small break LOCA analysis applicable to Fort Calhoun Station Unit No. 1 when a revised licensing model is approved consistent with the schedule prescribed in the letter from Darrell G. Eisenhut, dated May 7, 1980.

Automatic Trip of RCP's

The District is participating with the CE Owners Group for resolution of the automatic trip of RCP's issue.

Natural Circulation

The CE Owners Group has submitted predictions for the S-07-10B and L3-1. We believe these analyses show the adequacy of the small break LOCA model's handling of natural circulation phenomenon.

PORV Failures in CE Plants

The District will participate in the CE Owners Group's production of a report on PORV failure reduction and will design and install an automatic PORV isolation system if required by the report.

Other Items

The District is addressing other "Final Recommendations of the B&O Task Force" in accordance with our response to the letter from Darrell G. Eisenhut to All Operating Reactor Licensees, dated May 7, 1980.

Question 6

Provide references for NRC's safety evaluation for the quoted code versions which were used in this analysis.

Response

The computer code versions used in the small break ECCS analysis, listed in Section 6 of Reference 1, are presented again below along with the appropriate reference to the NRC's safety evaluation report.

<u>Code</u>	<u>Version No.</u>	<u>NRC Approval Letter</u>
CEFLASH-4AS	77019	Reference 6
STRIKIN-II	77036	Reference 7
COMPERC-II	74223	Reference 8
PARCH	77004	Reference 9

Question 7

Provide verification that the current core is identical to that assumed in the Cycle 6 reload analyses, e.g., no substitutions or burnup differences.

Response

The operating Cycle 6 core is within the bounds of the assumptions used in the Cycle 6 reload analysis.

Question 8

Provide verification that (1) there are no unusual anomalies associated with the CE, high burnup, demonstration assembly (e.g., power spike), (2) the demo assembly burnup will remain less than 40,000 EFPH, and (3) the reactor coolant system chemistry has been and will continue to be "normal".

Response

The Cycle 6 performance evaluation for the high burnup demonstration assembly was provided in a letter from W. C. Jones to Robert W. Reid, dated February 12, 1980. The demonstration assembly has no unusual anomalies. The assembly burnup will remain within the bounds assumed in the analysis. The reactor coolant system chemistry will be maintained within the Technical Specification limits.

REFERENCES

1. Fort Calhoun Small Break ECCS Performance Results at 1530 MWt, Letter, W. C. Jones to Robert A. Clark, May 21, 1980.
2. CENPD-133, Supplement 3, "CEFLASH-4AS, A Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-Coolant Accident", January 1977.
3. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", April 1977.
4. CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core", April 1974.
5. CENPD-138, Supplement 2, "PARCH, A Fortran IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", January 1977.
6. Letter, Karl Kniel to A. E. Scherer entitled, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P", September 27, 1977 (attached).
7. Letter, Robert L. Baer to A. E. Scherer entitled, "Evaluation of Topical Report CENPD-135, Supplement 5", September 6, 1978 (attached).
8. Letter, Olan D. Parr to F. M. Stern entitled, "NRC Staff Review of the Combustion Engineering ECCS Evaluation Model", June 13, 1975 (attached).
9. Letter, Karl Kniel to A. E. Scherer entitled, "Evaluation of Topical Report CENPD-138, Supplement 2-P", April 10, 1978 (attached).
10. CENPD-137, Supplement 1, "Calculative Methods for the C-E Small Break LOCA Evaluation Model", January 1977.