UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 Introduction

By applications dated April 30, 1981, April 30, 1982 and July 13, 1982, and supplemental information dated April 20, 1982 and June 24, 1982, Carolina Power and Light Company (the licensee) requested amendment to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (the facility). The amendment requests consist of:

a. Appendix A Technical Specifications (TSs) changes resulting from the analysis of the Cycle 9 reload.

b. Contined approval to operate through Cycle 9 at reduced power.

c. Appendix A Technical Specification (TS) changes resulting from surveillance requirements for ECCS Motor Operated valves.

d. Approval of an Operating License change for steam generator inspection and surveillance.

Carolina Power and Light Company (CP&L), proposes operation of HBR-2 at reduced power, primary temperature flow. Table 1 presents a comparison of rated power and reduced power major plant parameters. The licensee's new analysis was performed by Exxon Nuclear Company

(ENC). The program of reduced temperature, flow and power is proposed to improve the operating conditions of the steam generators, and to allow up to 20% tube plugging. This program is expected to result in a maximum power output of 85% of rated power.

TABLE 1

	Rated Conditions	Cycle #9
Power	2300 Mwt	1955 Mwt
Primary Flow	89965 gpm/loop	82700 gpm/loop
T_{ave}	$575^{o}F$	$537^{o}F$
Primary Pressure	2250 psia	2250 psia
Steam Generator Pressure	800 psig	580 psig

Operation at reduced power and temperature was started during Cycle #8. HBR-2 licensing Amendment No. 61, issued by NRC on November 13, 1981, consisted of changes to the Operating License and Technical Specifications to allow HBR-2 operation at reduced power, primary temperature and flow for the remainder of Cycle #8. This amendment stipulated that if the licensee wished to continue operation at reduced power, primary temperature, and flow after refueling, a detailed transient and accident analysis would have to be submitted for NRC review and approval. The licensee submitted this analysis in Reference (1). Reference (1) includes evaluation of the following anticipated operational occurrences (AOOs) and accidents:

AOO's - Uncontrolled rod withdrawal
- Three reactor coolant pump coastdown
- Loss of external load
- Excess load

Accidents - Loss of Coolant Accident (LOCA)
- locked rotor
- steam line break (SLB)

The following transients and accidents were not initially reanalyzed: startup of an inactive loop, loss of feedwater, loss of A.C. power, chemical and volume control system (CVCS) malfunction, steam generator tube rupture (SGTR) and reduction in feedwater enthalpy accident. Of the above, startup of an inactive loop and reduction in feedwater enthalpy were analyzed in Reference (2) and in the FSAR under full power conditions and showed acceptable consequences. Based on our request, the licensee provided information which discussed the consequences of the following transients at reduced power, temperature and flow: SGTR, CVCS malfunction, loss of offsite A.C. power, and loss of normal feedwater. The SGTR and loss of normal feedwater transients are evaluated in their respective sections. ENC has further indicated that the CVCS malfunction transient consequences are bounded by the rod withdrawal event and that the consequences of the loss of offsite A.C. power event are bounded by the 3 RCP coastdown transient with regard to minimum DNBR and loss of load event with regard to peak pressure. We conclude that operation at reduced power will not adversely affect the consequences of these transients.

2.0 Discussion and Evaluation

2.1 Fuel Design

The reload core design for Cycle 9 utilizes gadolinia as a burnable poison. The reload analysis makes use of gadolinea fuel properties described in Exxon topical report, XN-NF-79-56, which has been reviewed and approved by the NRC staff. Carolina Power and Light has stated that the gadolinia concentration in the fuel will be within those limits specified in our review of XN-NF-79-56. We find this to be acceptable.

2.1.1 Fuel ECCS Analysis

The staff has been generically evaluating three fuel material models that are used in ECCS analyses. Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage. The staff has (a) discussed its evaluation with vendors and other industry representatives (Ref. 3), (b) published NUREG-0630 (Ref. 4), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with the ECCS Acceptance Criteria of 10 CFR Part 50.46 if the NUREG-0630 correlations were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (Refs. 5 and 6) to offset penalties incurred due to the use of the NUREG-0630 correlations.

Although Exxon has submitted a new ECCS evaluation model (EXEM/PWR, see Ref. 7) that incorporates revised materials models (Ref. 8), the NRC review of the new ECCS evaluation model has not been completed and this model has not been used for the HBR LOCA analysis. Hence, in accordance with the requirements discussed in the preceeding paragraph, the HBR analysis has been augmented by a supplemental ECCS assessment that addresses the predicted effect of NUREG-0630 correlations on the HBR analysis.

In Reference 9, CP&L has provided this supplemental ECCS assessment. For operation at reduced temperature and power, the ECCS analysis of the HBR limiting double-ended cold-leg quillotine break at beginning-of-life conditions predicts reflood rates greater than 1 inch per second and peak cladding temperature (PCT) occurring on the burst node. Hence, reflood heat transfer calculations are performed with the FLECHT correlation and cladding rupture and burst strain models impact PCT analyses only at the burst node.

Exxon has performed sensitivity calculations using the ENC WREM-II PWR and EXEM/PWR ECCS evaluation models. The latter EM is the most recent and is currently under NRC review. It contains (a) cladding models that are slightly modified versions of the NUREG-0630 correlations and (b) various other model revisions such as cladding radiation heat transfer. Exxon has found that, with the new EM, an analysis of a burst-node-limited plant that uses FLECHT heat transfer correlations (such as HBR) will exhibit reduced LOCA PCTs compared with the old EM primarily because of the beneficial effect of the new radiation heat transfer model, which delays fuel rod rupture thus resulting in less cladding inner surface oxidation and the concurrent reduction in heat production associated with the metal-water reaction.

We thus conclude that the inclusion of the NUREG-0630 correlations into the HBR ECCS analysis would not result in predictions that exceed the ECCS Acceptance Criteria. Therefore, the issue of cladding swelling and rupture is resolved for HBR.

2.2 Nuclear Design

Physics parameters remain essentially unchanged from those for previous cycle (Cycle 8) operation at reduced primary coolant temperature and, therefore, are acceptable. However, more detailed information regarding transient and accident analyses was reviewed.

Transient analyses for the uncontrolled control rod withdrawal events from hot zero power and from 1955 MWt were presented in XN-NF-82-18.(Ref. 1). These were reviewed and found to be acceptable. The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from low power and full power conditions have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such control rod assembly accidents.

The staff also requested additional information on the control rod ejection accident which was supplied (Ref. 9). The assumptions and calculational techniques used are the same as those which have previously been evaluated by the staff and found to be acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The radial peak power value at BOC is less than that calculated in the reference analyses and is, therefore, acceptable. However, at EOC conditions, a peak radial power about 8 percent above the reference calculation peaking factor prior to ejection is calculated. This 8 percent increase, however, is more than offset by the 15 percent reduction in reactor operating power for Cycle 9. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained during a control rod ejection transient.

2.3 Thermal-Hydraulics

To support the reduced temperature program, the licensee has performed a review of anticipated operational transients at the proposed operating conditions and reactor protection system setpoints. The thermal-hydraulic calculations for the steady-state conditions at the reduced power and coolant temperature have shown about a 65 percent increase in MDNBR as compared to the rated full load operating conditions. Based on this substantial increase in thermal margin, the licensee concludes that the anticipated operational transients will satisfy the Specified Acceptable Fuel Design Limits (SAFDLs) since the changes in MDNBR during these transients will not be greater than those previously evaluated for rated full power. The staff agrees with this conclusion although additional information for certain reactivity initiated transients (discussed below) were requested.

For large steam line break analysis, the modified Barnett critical heat flux (CHF) correlation (Ref. 10) is employed for DNBR calculation. However, no DNBR limit, which will ensure avoidance of a fuel rod experiencing DNB with 95 percent probability at 95 percent confidence level, was described in XN-NF-82-18 (Ref. 1). In a telecommunication (Y. Hsii of NRC and J. C. Chandler of ENC on June 9, 1982), Exxon indicated the DNBR limit for the modified Barnett correlation was 1.135. This 95/95 DNBR limit was developed from the CHF data presented in the Appendix A of Reference 10using the Non-Parametric Tolerance Limit Method (Ref. 11). Our evaluation has found that the modified Barnett correlation with a DNBR limit of 1.135 is acceptable for the steam line break analysis based on the following observations: (1) The non-parametric method is a distribution-free tolerance limit determination method with no assumption of normal distribution regarding the measured-to-predicted CHF ratio data. Therefore, it is a proper method for determining the DNBR limit. (2) The modified Barnett correlation has been approved in 10 CFR Part 50, Appendix K as an acceptable CHF correlation for LOCA analysis. We conclude that it is also acceptable for the steam line break transient analysis where the primary system pressure falls within the pressure range of 150 to 725 psia of the modified Barnett correlation. (3) The DNBR limit of 1.135 is determined with 95/95 probability/confidence level from the existing CHF data described in Reference 10.

3.0 Anticipated Operational Occurrences

3.1 Three Rector Coolant Pumps (RCP) Coastdown

This analysis assumed loss of power to all three RCPs at 1955 Mwt power level, beginning of cycle reactor kinetics coefficient, and reactor trip on low flow signal (more conservative than the more realistic assumption of reactor trip due to bus undervoltage or underfrequency). A multiplier of 0.8 was applied to the Doppler coefficient for conservatism. The pressurizer was assumed to be in automatic control with pressurizer spray available. While this takes credit for non-safety grade equipment, it is more conservative with regard to DNBR prediction, since actuation of the pressurizer spray results in a lower DNBR. The minimum DNBR was 2.58 at 3.5 seconds. The peak primary pressure is bounded by the loss of external load event (see item 3 below). We conclude that this analysis is acceptable.

3.2 Excess Load

This analysis assumed increase in turbine load causing a power mismatch between reactor power and steam generator demand. A 10% step increase in rated turbine load was analyzed at an initial power of 1955 Mwt, end of core life, with no automatic control rod or pressurizer control assumed. Core power reached 2115 Mwt after 42 seconds. Minimum DNBR was 2.79 at 51 seconds. Both primary and secondary pressure decreased. We conclude that this analysis is acceptable.

3.3 Loss of Load

This analysis assumed a tubine trip without a direct reactor trip, an initial power level of 1955 Mwt at beginning of core life, thus providing a positive moderator coefficient. For conservatism, a multiplier of 0.8 was applied to the Doppler coefficient. No credit was taken for automatic reactor control, steam dumps and tubine bypass. However, the initial reanalysis assumed that pressurizer spray and the power relief valves (PORVs) were operational. This assumption was conservative for DNBR prediction because of lower pressures as a result of pressurizer spray and PORV actuation, but not for predicting peak pressure. Reactor trip on high pressure occurred in 12.5 seconds, and primary pressure peaked at 2460 psia in 14 seconds. By comparison PORV actuation is at 2335 psig and primary safety valve actuation at 2485 psig. The minimum DNBR was 2.91.

Based on our request this transient was reanalyzed for peak primary system pressurization (Ref. 9). In this reanalysis, the PORVs and pressurizer spray were assumed inoperable. The predicted peak primary pressure was 2585 psia. The primary safety valves would be actuated. There was no decrease in DNBR from its original value. Therefore, we conclude that this analysis is acceptable.

3.4 Loss of Normal Feedwater

This event as analyzed in the original FSAR and was not reanalyzed in references (1) and (2). The FSAR analysis indicated that for rated power conditions T(average) peaked at 605°F approximately 1500 sec after initiation of the transient, and that there was no water relief from the pressurizer relief or safety valves. Based on our request for additional information, the licensee provided an estimate of the results of this transient during reduced power and primary temperature operation, which predicts a maximum T(average) of 608°F, and pressurizer safety valve actuation, resulting in expulsion of 140 cubic feet of primary fluid. The time after transient initiation for occurrence of these events was not given.

These analyses were based on the assumptions of a reactor trip on steam flow/feedwater flow mismatch coincident with steam generator low water level or on low-low steam generator level, natural circulation in the primary loops, one auxiliary feedwater pump starting at one minute and delivering 300 gpm to two steam generators, no credit for steam dump valves, and steam generator safety valve actuation. These assumptions are conservative. The licensee stated that there would be no fuel damage since about 850 cubic feet of liquid remains above the core, and that sufficent auxiliary feedwater capacity exists to remove decay heat. Results of loss of mainfeedwater analyses for other Westinghouse plants at full power conditions also indicate that there is no DNBR problem. The licensee has further indicated that the DNBR for this event is bounded by the DNBR for the 3 reactor coolant pump coastdown transient (See Section 3.0). We conclude based on our review of other plants as well as the H. B. Robinson 2 submittal, that DNBR will remain acceptable.

However, since the licensee's analysis is unrealistically conservative and may mask other effects in the transient, we require that the licensee perform a more detailed analysis for this transient. The results of this analysis should include plots of T(average), primary and secondary pressure versus time for the full extent of the transient, and the value for the minimum DNBR attained. These results should be submitted to NRC by October 31, 1982.

4.0 Accidents

4.1 LOCA

A new LOCA ECCS analysis for only the limiting break was performed for the HBR-2 reduced power and primary temperature operation. The licensee states that the analysis was performed in accordance with 10 CFR Part 50, Appendix K, for the limiting double ended cold leg guillotine break at beginning of life fuel conditions. Previous analyses showed this to be the limiting break with regard to peak cladding temperature (PCT) (see References 12 and 13). The EMC WREM-IIA model was utilized. A discharge coefficient (C_D) of 0.8 was assumed, as previous analysis had shown this to be conservative. (see Reference 14). Loss of offsite power was assumed.

The ENC analysis identifies a number of detrimental effects for the reduced power, temperature and flow operation as compared to rated conditions for the LOCA consequences. These included: reduced heat transfer during blowdown because of decreased core flow; a slower core power decay due to reduced voiding; reduced reflood rates due to lower containment pressure; longer blowdown times because of reduced saturation pressures with lower pressures earlier in the blowdown, which in turn result in earlier accumulator injection and flow for a longer time during blowdown, with consequent greater loss of accumulator inventory, since 10 CFR Part 50 Appendix K requires all ECCS coolant injected during blowdown to be assumed lost. Nevertheless, the reduction in linear heat generation rate associated with the 15% reduction in power more than offsets these detrimental effects and results in a PCT of 2077^{O}F compared with a PCT of 2185^{O}F for a LOCA at full power and at rated temperature and flow. The maximum local metal-water reaction is 6.05% and total core-wide metal-water reaction is less than 1%, thus meeting the requirements of 10 CFR Part 50.46.

Based on our request, the licensee provided information (Ref. 9) which indicates that consideration of the cladding swelling and rupture model in NUREG-0630 would not adversely affect prediction of PCT (discussed in Section 2.1.1). We conclude that the LOCA analysis at reduced power and temperature is acceptable.

4.2 Locked Rotor

This analysis assumes three loop operation at 1955 Mwt, with instantaneous seizure of one RCP. The reactor is tripped by the resulting low flow signal. The feedwater pumps were assumed to trip with the reactor, but offsite power is retained and continued operation of the intact RCPs is assumed. Beginning-of-cycle reactor kinetics coefficients are assumed. A 0.8 multiplier is applied to the Doppler coefficient for conservatism. A 0.95 multiplier was applied to the DNBR to account for assymetric core flow because of loop flow differences due to steam generator tube plugging. Based on these assumptions, the minimum predicted DNBR is 2.19 and peak primary pressure is 2321 psig. We conclude that this analysis is acceptable.

4.3 Steam Line Break (SLB)

The SLB was reanalyzed for the most severe case i.e., an SLB inside containment at end of core life and at hot zero power conditions, corresponding to a core average temperature of 530°F. At this time the steam generator secondary side inventory is at a maximum, prolonging the duration and increasing the magnitude of the primary loop cooldown. For additional conservatism, offsite power is assumed available, the most reactive control rod is assumed to be stuck out of the core, the break is assumed to occur at the steam generator with the fewest plugged tubes and blowdown occurs also from the other two steam generators until closure of the main steam isolation valves.

The analysis shows very rapid loss of both primary and secondary pressure when compared to other SLB analyses on similar PWRs. The faulted steam generator is almost completely depressurized in 1-2 seconds and primary pressure decreases to about 250 psia in 50 seconds. In addition the licensee's analysis shows that the core returns to power at 7.5 seconds. These results appear to be inconsistent with analyses for other Westinghouse plants which show a much slower depressurization of the faulted steam generator and considerably higher minimum primary pressure. The peak power reached is approximately 940 Mwt at 43 seconds, after which boron addition terminates the power increase. The minimum critical heat flux (CHF) is calculated to be 1.19 at the time of peak core heat flux, utilizing the modified Barnett CHF correlation (discussed in Section 2.3). This value appears adequate based on a minimum acceptable CHF of 1.135. Discussions with the licensee indicated that the SLB model utilized does not consider asymmetric core temperatures, nor the mass input and RCS cooldown due to accumulator actuation or SIS input. The analysis does assume the boron addition from high pressure SIS to shutdown the reactor after its return to criticality due to the cooldown. The model utilized appears to provide conservative values and the resulting CHF appears acceptable. Therefore, we conclude, based on our review of MSLBs at other W plants and our review of the H. B. Robinson information, that the consequences of a MSLB at reduced power and temperature will not result in unacceptable fuel performance. However, since the licensee's analysis is excessively conservative and does not assume the mass input from the SIS, the analyses may mask important system effects. Therefore, we require that the licensee provide additional information that justifies the adequacy and conservatism of the model utilized in the SLB analysis, prior to the next refueling.

4.4 Steam Generator Tube Rupture (SGTR)

This event was analyzed in the original FSAR and was not reanalyzed in Reference (1) and (2). Based on our request, the licensee provided information which indicates that, despite the larger initial primary to secondary pressure differential, total primary to secondary leakage is estimated to be 4000 lbs. less for reduced power operation than for full power operation, and thus the consequences of this accident would be less severe. The consequences of this accident at rated conditions was previously reviewed and found acceptable. We conclude that the consequences of this event at reduced power conditions are acceptable.

5.0 Technical Specifications

5.1 Reduced Temperature Program

For the reduced temperature program, the licensee proposes changes to the technical specifications (Ref. 15). These changes include:

5.1.1 The peak F_Q (including uncertainties) assumed for Cycle 9 operation is revised to 2.32 at 85% of rated power. The revised F_Q limit of 2.32, corresponding to a linear heat generation rate of 11.8 KW/ft, is used in the LOCA ECCS analysis for reduced temperature operation and results in acceptable consequences. For additional analyses of the more limiting transients for reduced temperature operation, a more conservative value of 2.55 is used, also with acceptable consequences. The revised F_Q limit is bounded by the value used in the LOCA and other limiting transient analyses and is, therefore, acceptable.

5.1.2 The terms "rated power", "full power", "rated values", and "design values" are redefined under the reduced temperature program with power operation at 1955 MWt. The identification of the power level that various Limiting Conditions of Operation (LCO) are related to during the reduced temperature operation is primarily for clarification and is acceptable.

5.2 Additional Technical Specification Change

By application dated April 30, 1981, the licensee requested a change in the Technical Specifications to require specified surveillance of the Emergency Core Cooling System (ECCS) Motor Operated Valves which is required as a result of modifications to the ECCS electrical control circuits.

These changes were requested by our letter dated March 9, 1981 which suggested acceptable surveillance. The licensee responded to our request and used our suggested surveillance. Therefore, this change is acceptable.

6.0 Licensing Condition

By letter dated July 13, 1982, the licensee requested a modification to the Operating License Condition 3.I.a, b, c & d.

6.1 Steam Generator

As a result of a high level of stress corrosion cracking activity above the tubesheet area observed during August 1981, license conditions were imposed for the balance of Cycle 8 operation which included periodic primary to secondary hydrostatic tests, and more stringent limits on allowable primary to secondary leakage. The eddy current inspection results performed during the current outage indicates that reduced temperature operation since November 1981 has been successful in sharply reducing stress corrosion cracking activity above the tubesheet. The licensee plans to continue reduced temperature operation (T_{av} = 537°F) during the next cycle. For this reason, the staff has concluded that there is reasonable justification for not reimposing the license condition for periodic hydrostatic tests during the next operating cycle. Stress corrosion cracking and intergranular attack continues to be active within the tubesheet crevice region. However, the narrow tube to tubesheet crevices or gaps severely limit the potential for any high leakage such as could occur as a result of a rupture in free span portions of tubing (i.e., above the tubesheet). The licensee has proposed to continue the license condition for reduced limits on primary to secondary leakage which were imposed for the balance of Cycle 8 operation following August 1981.

Eddy current inspections have indicated an accelleration of phosphate wastage corrosion during the past operating cycle. By letter dated July 13, 1982, the licensee has proposed a licensing change which would require shutdown of H. B. Robinson within 6 EFPM of restart from the current outage for additional steam generator inspections to ensure that further progression of wastage does not become excessive. The licensee provided the staff with the eddy current inspection results, eddy current error estimates, and projected corrosion rates for the next cycle of operation to justify six months operation. This information is still being reviewed by the staff. However, based upon our preliminary findings, we have concluded that H. B. Robinson can be operated safely for at least three EFPM in a manner reasonably consistent with the criteria (per Regulatory Guide 1.121) which the staff generally employs for this type of evaluation. We plan to complete our evaluation of the licensee's proposed six EFPM operating interval by September 3, 1982. Operation beyond three EFPM to six EFPM as proposed by the licensee will be subject to approval by the staff.

7.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of enviornmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 23, 1982

References

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2. ENC Report XN-75-14 "Plant Transient Analysis of the H. B. Robinson Unit 2 for 2300 Mwt" July 1975.

3. R. P. Denise (NRC) memorandum for R. J. Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979.

4. D. A. Powers and R. O Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980.

5. D. G. Eisenhut (NRC) letter to All Operating Light Water Reactors, November 9, 1979.

6. H. R. Denton (NRC) memorandum for Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979.

7. "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon report XN-NF-82-20, March 1982.

8. "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon report XN-NF-82-07, March 1982.

9. S. Zimmerman (CP&L) letter to S. Varga (NRC), June 24, 1982.

10. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux For Water in the Pressure Range 150 to 725 PSIA," Idaho Nuclear report IN-1412, July 1970.

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12. ENC Report XN-75-57 "HBR-2 LOCA Analyses Using the ENC WREM Based PWR ECCS Evaluation Model" October 1975.

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14. ENC Report XN-76-54 "LOCA Analyses for HBR-2 Using WREM Based PWR ECCS Evaluation With Reduced LPSI Flow, Steam Generator Plugging and Increased Upper Head Temperature" December 1976.

15. B. J. Furr (CP&L) letter to S. A. Varga (NRC), April 30, 1982.