Docket No. 50-261

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Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, REGARDING REMOVAL OF OPERATING RESTRICTIONS FOR SINGLE HIGH PRESSURE INJECTION PUMP OPERATION (TAC NO. 68072)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 119 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated May 7, 1988, and supplemented by letters dated May 16 and May 20, 1988.

The amendment removes the restriction limiting operating power to 1380 MWt with two operable safety injection pumps and increases the power peaking factor (Fq) to 2.32 from a value of 2.26. The amendment allows operation at a steady state reactor core power level not in excess of 2300 MWt with two safety injection pumps operable.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance has been forwarded to the Federal Register for publication.

Sincerely,

/S/

Ronnie H. Lo, Sr. Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 119 to DPR-23
2. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000261
P PDR

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:	
NAME	: PAnderson	: RLo:clh	: EAdensam	:	:	:	
DATE	: 6/16/88	: 6/17/88	: 6/20/88	:	:	:	

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AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-23 - ROBINSON, UNIT 2

Docket File
NRC PDR
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L. Lois (8E23)
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated May 7, 1988, as supplemented May 16, and May 20, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 119 , are hereby incorporated in the license. Carolina Power & & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 20, 1988

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	RLo:clh	MYoung	EAdensam			
DATE	6/15/88	6/15/88	6/16/88	6/20/88			

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ATTACHMENT TO LICENSE AMENDMENT NO. 119

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3.3-2	3.3-2
3.3-3	3.3-3
3.10-2	3.10-2
3.10-2a	
3.10-3	3.10-3
3.10-3a	
3.10-4	3.10-4
3.10-4a	
3.10-5	3.10-5
3.10-5a	
3.10-22	3.10-22

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated.

c. Two safety injections pumps are operable, each capable of automatic initiation from a separate emergency bus.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All essential features including valves, interlocks, and piping associated with the above components are operable.

g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position:

Valves	Position
MOV 862 A&B	Open
MOV 864 A&B	Open
MOV 865 A,B,&C	Open
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed

h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position.

i. Power operation with less than three loops in service is prohibited.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. One accumulator may be isolated for a period not to exceed four hours.

b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the remaining safety injection pump is demonstrated to be operable prior to initiating repairs.

c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other residual heat removal pump is demonstrated to be operable prior to initiating repairs.

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions $\geq$ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < 4.64 \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < 1.65\ (1 + 0.2(1-P)\ )$

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q(Z)$ including the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}$ including a 1.04 measurement uncertainty factor. K(Z) is based on the function given in Figure 3.10-3, and Z is the axial location of F_Q.

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.2 $F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

where V(Z) is defined in Figure 3.10-4 which corresponds to the target band and $P > 0.5$.

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)}\right] -1 \right] \times 100\%$$

c) Comply-with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

$$\text{APL} = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q(Z)$, including the engineering factor $F_Q^E = 1.03$ and the measurement uncertainty factor $F_u^N = 1.05$ at the time of target flux determination from a power distribution map using the movable incore detectors. V(Z) is the variation function defined in Figure 3.10-4 which corresponds to the target band. K(Z) is the function defined in Figure 3.10-3.

The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$[F_j(Z)\ S(Z)]_{max} \leq \frac{2.103/P}{\bar{R}_j\ (1 + \sigma_j)}$$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. $\bar{R}_j$ for thimble j, is determined from core power maps and is by definition:

$$\bar{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qj}}{[F(Z)_{ij}\ S(Z)]_{max}}$$

F_{Qj} is the value obtained from a full core map including S(Z), but without the measurement uncertainty factor F_u^N or the engineering uncertainty factor, F_Q^E. The quantity $F(Z)_{ij}\ S(Z)$ is the measured value without inclusion of the instrument uncertainty factors F_Q^a. Those uncertainty factors, $F_u^N = 1.05$, $F_Q^a = 1.02$, as well as the engineering factor $F_Q^E = 1.03$, have been included in the limiting value of 2.103/P.

c. σ_j is the standard deviation associated with the determination of $\bar{R}_j$.

d. S(Z) is the inverse of the K(Z) function given in Figure 3.10-3.

This limit is not applicable during physics tests and excore detector calibrations.

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F_{\Delta H}^N$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

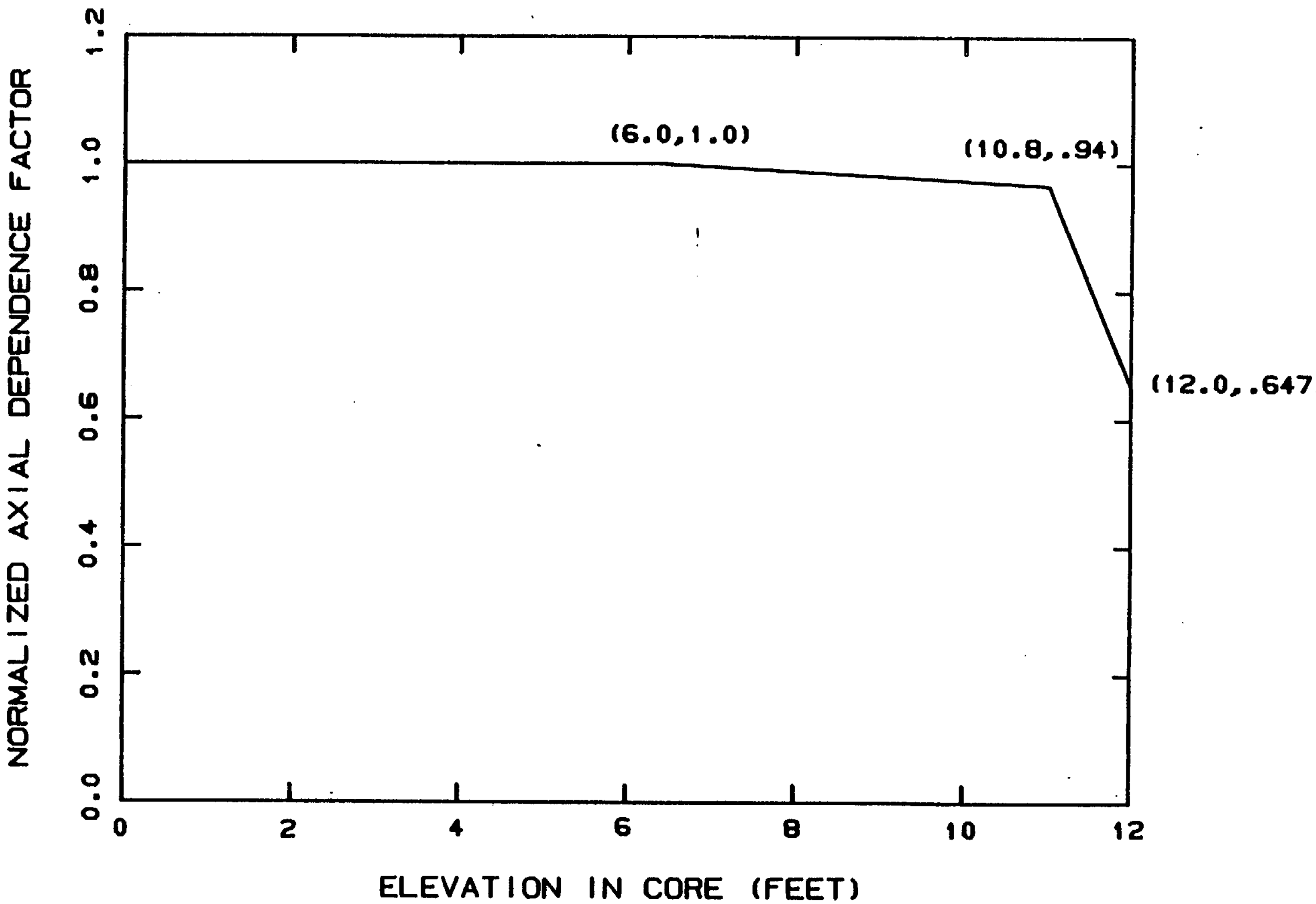

FIGURE 3.10-3 NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION (PEAK Fq = 2.32)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

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

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated May 7, 1988, and supplemented by letters dated May 16, and May 20, 1988, the Carolina Power & Light Company (CP&L) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment would change the Technical Specifications (TS) by removing the restriction limiting operating power to 1380 MWt with two operable safety injection (SI) pumps and increase the power peaking factor (Fq) to 2.32 from a value of 2.26. The amendment would allow operation at a steady state reactor core power level not in excess of 2300 MWt with two SI pumps operable.

2.0 EVALUATION

The limitations on power level and power peaking factor were imposed by Amendment No. 115, dated March 7, 1988 because of concerns regarding single failure vulnerability of the SI system (Refs. 1-3). Early in 1988, CP&L discovered that at least one postulated single failure event exists which could result in the loss of the ability to automatically start two high head safety injection pumps during a loss-of-coolant-accident (LOCA) event. Upon examination of the possible failure events it was postulated that only one high head safety injection pump would be available during a LOCA. The licensee has performed new LOCA analyses assuming only one high head safety injection pump. The request for removal of the operating limitations is based on the results of those analyses.

The small break LOCA analysis was performed with the NRC-approved NOTRUMP methodology (Ref. 4, 6). The licensee also submitted an updated large break LOCA analysis performed by Advanced Nuclear Fuels Corporation (ANF) with the EXEM/PWR, ECCS evaluation package (Ref. 5). The main input changes consist of a revised single high head safety injection pump flow and a revised axial power distribution.

2.1 Revised High Head Safety Injection Pump Flow

Previous ECCS analyses had assumed a value of high head injection flow based upon very conservative design assumptions. In the revised analyses, the Westinghouse Corporation calculated pump delivery based on as-built piping and a minimum pump curve based on system test performance degraded by 5% of the design values. The revised estimate results in greater flow. Although this is less conservative from a LOCA analysis perspective than the previous assumptions, the staff considers it acceptable because the use of the plant-specific, as-built piping configuration and actual pump test data provide assurance that the flow has been correctly modeled.

2.2 Power Shape Selection

Appendix K to 10 CFR Part 50 states that "A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime shall be studied and the one selected should be that which results in the most severe calculated consequences, for the spectrum of postulated breaks and single failures analyzed." ANF, the fuel supplier for the Robinson plant, supplied a data base of about 120 power shapes, generated for several core power levels and representing beginning, middle, and end of cycle conditions. Power shapes were obtained from bounding load follow cases under the ±5% I operating band. Highly peaked power distributions were obtained by using increased control rod worths. ANF stated that "the data base represents a good census of limiting small break LOCA power shapes." The method followed is the same as in the establishment of limiting power distributions for determining axial offset limitations. Of particular interest is the power distribution in the upper 2-3 feet of the core which could, under conditions of a small break LOCA, become uncovered and therefore, limiting with respect to peak clad temperature. For this part of the core power distribution, the described method includes the limiting shape from a broad range of possible shapes. It is then scaled to the K(z) axially dependent power peaking limit distribution, thus assuring a bounding conservative distribution.

2.3 Small Break Loss-of-Coolant-Accident (SBLOCA)

The SBLOCA analyses were performed using the staff approved NOTRUMP and LOCTA-IV codes (Ref. 7). Inputs included the flow and power shape assumptions described above and a steam generator tube plugging level of 5 percent. A spectrum of three break sizes of 1.0, 1.5 and 2.0 inches in equivalent diameter were analyzed. The 1.5 inch break was found to be limiting.

For the limiting break, the peak clad temperature was estimated to be 2,004°F, the maximum local Zr-H_2O reaction was 8.59% and the total Zr-H_2O reaction was less than 0.30%.

The methodology described above for the small break LOCA satisfies the requirements of Appendix K to 10 CFR Part 50 and the results satisfy the limits of 10 CFR 50.46. The analyses are, therefore, acceptable.

2.4 Large Break LOCA

This analysis was performed to support an increase in the enthalpy rise factor $F_{\Delta H}$ from the reduced value of 1.65 to 1.70. The calculation assumed only one high head safety injection pump operating, used the existing power peaking K(z) limit and a double ended cold leg guillotine break with a discharge coefficient of 0.8. In addition a full ANF 15x15 fuel core was assumed, incorporating the 17x17 fuel cooling test facility data for the calculation of quench time and velocity, the carryover rate fraction and the heat transfer coefficient during reflood. Scaling of the 17x17 fuel assembly data to the 15x15 Robinson fuel is described in Reference 8, which has been approved by the NRC. This is the first time that the ANF large LOCA methodology is applied in the Robinson plant. The power level was assumed at 2346 MWt i.e., 102% of the licensed power. Other assumptions include a total peaking factors of F_Q=2.32, $F_{\Delta H}$=1.70 and steam generator tube plugging of 6%. The calculations were performed using fuel rod conditions at the exposure when maximum stored energy occurs i.e., 1,800 MWD/MTU.

The methodology employed was ANF's EXEM/PWR ECCS evaluation model, which consists of the following components:

(a) RODEX2: for initial stored energy, fission gas release and gap conductance;

(b) RELAP4-EM: for system blowdown, hot channel blowdown and accumulator and safety injection flow calculations;

(c) CONTEMPT-LT/22: for containment back pressure (as modified to comply with CSB 6-1 Branch Technical Position);

(d) REFLEX: for system reflood; and

(e) TOODEE2: for the calculation of the final fuel rod heat up.

The above package has been approved by the NRC (Ref. 5).

The results of this analysis showed that all of the conditions of 10 CFR 50.46 were satisfied, i.e., peak cladding temperature of 1986°F, maximum local metal-water reaction of 2.53%, and the total core metal-water reaction is less than 1.0%.

2.5 Summary and Conclusions

Small break LOCA and large break LOCA analyses were performed with NRC approved methods and conservative input data for the H.B. Robinson Unit 2 plant. The purpose of the calculations was to demonstrate that the plant meets the 10 CFR 50.46 LOCA analyses criteria for F_Q=2.32, and $F_{\Delta H}$=1.70 with only one high pressure injection pump operating. This recalculation was prompted by the interim operating limitations of F_Q=2.26 and a power level of 60% of licensed power when it was discovered that under LOCA conditions possibly only one high pressure injection pump would be operational.

The results of the calculations demonstrated that the reactor meets all of the requirements in 10 CFR 50.46 and, thus, are acceptable.

The staff notes that the analyses assumed 5% and 6% steam generator tube plugging for the small and the large break LOCAs respectively. Therefore, if the steam generator plugging exceeds 5% (or 6%, for large break LOCA), it will be necessary for the licensee to demonstrate that the analyses remain valid for higher percentages of plugged tubes.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published (53 FR 22751) in the Federal Register on June 17, 1988. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The Commission has issued a Notice of Consideration of Issuance of Amendment of Facility Operating License and Opportunity for Hearing which was published in the Federal Register (53 FR 17996) on May 19, 1988. No petition to intervene or request for hearing has been filed on this action.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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 nor to the health and safety of the public.

Principal Contributors: Wayne Hodges
Lambros Lois

Dated June 20, 1988

REFERENCES

1. Letter from M. A. McDuffie, Carolina Power and Light Company to USNRC (NLS-88-111), dated May 7, 1988, and additional information dated May 20 and May 24, 1988.

2. Letter from Westinghouse Electric Corporation to Carolina Power and Light Company, "Justification for Startup and Operation of H.B. Robinson at 100% Power With One High Head Safety Injection Pump Available," CPL-88-538, dated May 5, 1988.

3. Chen, T.H., "H.B. Robinson Unit 2, Large Break LOCA/ECCS Analysis with an Increased Enthalpy Factor" Report ANF-87-159, dated November 24, 1987 and Supplement 1, dated April 28, 1988.

4. WCAP-10080-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code" by P. E. Meyer and J. Kornfilt Westinghouse Electric Corporation, August 1985.

5. Letter from D. M. Crutchfield (USNRC) to G. M. Ward Exxon Nuclear Company, "Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," dated July 8, 1986.

6. WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," N. Lee et al., Westinghouse Electric Corporation, August 1985.

7. WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analysis" by F.M. Bordelon et al., Westinghouse Electric Corporation, dated June 1974.

8. XN-NF-105(P), "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs" by W.V. Kayser, Exxon Nuclear Company, dated October 1985 and Supplement 1 dated January 1986.