March 7, 1988

Docket No. 50-261

DISTRIBUTION
See attached page

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. 115 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, REGARDING OPERATION OF PLANT BELOW 1380 MWt (TAC NO. 67198)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 115 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 February 24, 1988, as supplemented February 26, 1988 and March 1, 1988.

The amendment changes the Technical Specifications so that the plant would be permitted to be operated at less than 1380 MWt when only two safety injection pumps are operable. The amendment also requires NRC review and approval prior to operating above 1380 MWt.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing (Emergency Circumstances) will be included in the Commission's regular bi-weekly Federal Register notice.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 115 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

LA:PD21:DRPR
PAnderson
3/3/

PM:PD21:DRPR
RLo
3/4/

D:PD21:DRPR
EAdensam
3/4/88

Mr. E. E. Utley
Carolina Power & Light Company

H. B. Robinson 2

cc:
Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29535

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P.O. Box 629
Raleigh, North Carolina 27602

Mr. D. E. Hollar
Associate General Counsel
Carolina Power and Light Company
P.O. Box 1551
Raleigh, North Carolina 27602

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
Route 5, Box 413
Hartsville, South Carolina 29550

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street
Atlanta, Georgia 30303

Mr. R. Morgan
General Manager
H. B. Robinson Steam Electric Plant
Post Office Box 790
Hartsville, South Carolina 29550

Mr. Avery Upchurch, Chairman
Triangle J Council of Governments
100 Park Drive
Post Office Box 12276
Research Triangle Park, NC 27709

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P.O. Box 29520
Raleigh, North Carolina 27626-0520

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. 115
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 February 24, 1988, as supplemented February 26, 1988, and March 1, 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. 115 , 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/

Gus C. Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1988

LA:PD21:DRPR	PM:PD21/DRPR	OGC-B	D:PD21:DRPR	REG II	ADR2:NRR
PAnderson	RLo		EAdensam	LReyes	GLainas
3/3 /88	3/4/88	3/ /88	3/ /88	3/ /88	3/7/88

ATTACHMENT TO LICENSE AMENDMENT NO. 115

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

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages, as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3.3-2	3.3-2
3.3-3	3.3-3
3.3-4	3.3-4
3.10-2	3.10-2
3.10-2a	3.10-2a
3.10-3	3.10-3
3.10-3a	3.10-3a
3.10-4	3.10-4
3.10-4a	3.10-4a
3.10-5	3.10-5
3.10-5a	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. Three safety injections pumps are operable.*

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.

* With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, the reactor may be made critical; however, steady state reactor core power level shall not exceed 1380 megawatts thermal. Prior to exceeding 1380 megawatts thermal, NRC review and approval is required.

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 two safety injection pumps are 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.

* For reactor core power levels up to and including 1380 megawatts thermal, Specification 3.3.1.2.b shall be modified to read: "If one of the two automatically initiated safety injection pumps becomes inoperable, the reactor may remain in operation for a period not to exceed 24 hours, provided the remaining automatically initiated safety injection pump is demonstrated to be operable prior to initiating repairs."

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours, provided the other flow path(s) are demonstrated to be operable prior to initiating repairs. The hot leg injection paths of the Safety Injection System, including valves, are not subject to the requirements of this specification.

f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to four hours.

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 $P > 0.5$ (Note 1)

$F_Q(Z) < 4.64 \times K(Z)$ for $P \leq 0.5$ (Note 2)

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

Note 1: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$F_Q(Z) \leq (2.26/P) \times K(Z)$ for $P > 0.5$

Note 2: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$F_Q(Z) < 4.52 \times K(Z)$ for $P \leq 0.5$

where P is t fraction of rated power (2300 ( ) 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.

* 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.

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:

$$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5 \quad \text{(Note 3)}$$

$$F_Q(Z) < 4.64 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5 \quad \text{(Note 4)}$$

Note 3: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) \leq \left(\frac{2.26}{P}\right) \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P > 0.5$$

Note 4: With only twd safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$F_Q(Z) < 4.52 \left[\frac{K(Z)}{V(Z)}\right] \text{ for } P \leq 0.5$$

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:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.32}{P} \times K(Z)} \right] - 1 \right] \times 100\% \quad \text{(Note 5)}$$

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:

$$APL = \text{minimum over Z of } \frac{2.32 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\% \quad \text{(Note 6)}$$

Note 5: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$\left[\left[\text{Max. over Z of } \frac{F_Q(Z) \times V(Z)}{\frac{2.26}{P} \times K(Z)} \right] - 1 \right] \times 100\%$$

Note 6: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$APL = \text{Minimum of Z of } \frac{2.26 \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

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:

$$\left[F_j(Z)\ S(Z)\right]_{max} \leq \frac{2.103/P}{\bar{R}_j\ (1 + \sigma_j)} \quad \text{(Note 7)}$$

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:

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

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. σj 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.

Note 7: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, this formula shall be modified to read as follows:

$$\left[F_j(Z)\ S(Z)\right] \text{max.} \leq \frac{2.049/P}{\bar{R}_j\ (1 + \sigma_j)}$$

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

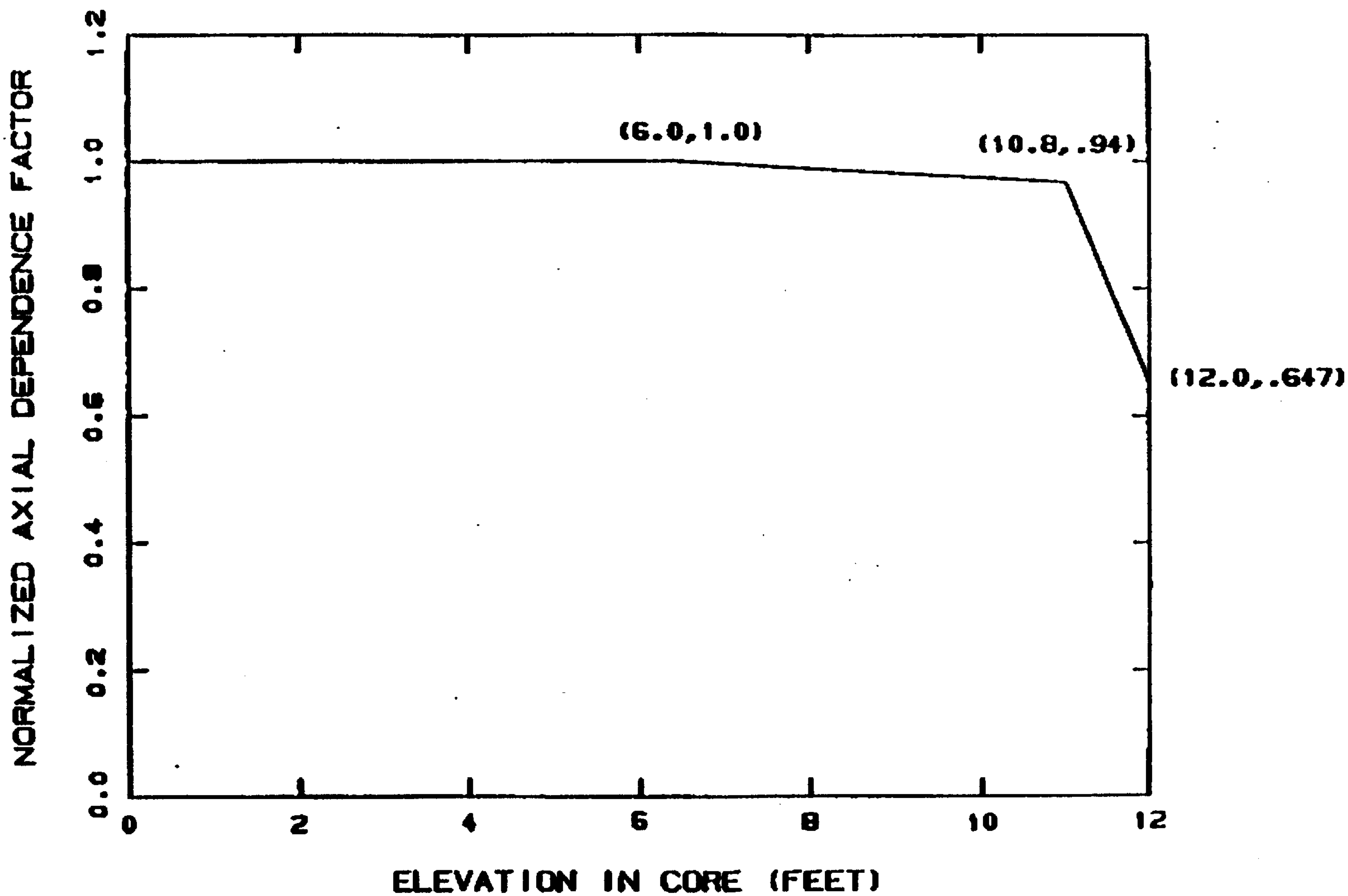

FIGURE 3.10-3 NORMALIZED AXIAL DEPENDENCE FACTOR FOR Fq VERSUS ELEVATION (PEAK Fq = 2.32)

Note: With only two safety injection pumps operable, each capable of automatic initiation from a separate emergency bus, peak Fq = 2.26.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 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 letters dated February 24, 26 and March 1, 1988, the Carolina Power & Light Company submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2, Technical Specifications.

The amendment changes the Technical Specifications to restrict the steady state reactor core power level to less than 1380 megawatts thermal (MWt) when only two safety injection pumps, each capable of automatic initiation from a separate emergency bus, are operable. The Technical Specification change also requires that, prior to exceeding 1380 MWt, NRC review and approval is required.

2.0 DISCUSSION AND EVALUATION

There are three safety injection (SI) pumps provided for H. B. Robinson, Unit 2. The current licensing basis for the plant takes credit for two of the three SI pumps to mitigate the design basis loss-of-coolant accident (LOCA). The SI pump A is powered from train A emergency bus E-1, while the pump C is powered from the redundant train B emergency bus E-2. The third "swing" pump B is normally energized from the E-1 through circuit breaker 22B. However, in the event of a failure of train A, the swing pump also can be energized from the train B emergency bus E-2 (through circuit breaker 29B) by utilizing an automatic bus transfer scheme. To prevent simultaneous closure of these circuit breakers (22B and 29B), interlocks are provided to ensure adequate independence of redundant trains.

With the present design configuration, the licensee discovered that a single failure of the diesel generator voltage regulator could result in loss of two of the three SI pumps (i.e., SI pump A and B). With only one SI pump the facility cannot satisfy the requirements of 10 CFR 50.46 while operating at full power. To eliminate this single failure vulnerability, the licensee has proposed a two SI pump configuration where two SI pumps (A&C) will be required operable under the proposed Technical Specification change, one SI pump on each train. In addition, the licensee will be required to limit the operating power level to 60% of the rated power. At this reduced power level, the licensee's LOCA analysis has determined that only one SI pump (assuming a single failure of the other SI pump) will be sufficient to mitigate the LOCA. Thus, the modification (Mod. 951) requires removing the auto start capability of SI "swing" pump B and opening of the SI pump B breaker 29C. The interlocks between circuit breakers (22B and 29B) and the auto sequences of two SI pumps (A&C) will remain unchanged. Although not required for the design basis accident, the SI pump B could be manually loaded and powered from either emergency bus.

To determine the allowable power level for the plant with only one SI pump for ECCS performance, after the assumption of a single failure, the licensee evaluated the depressurization events which may need the SI flow to mitigate the consequence during transients. The transients evaluated were: (1) inadvertent operation of a steam generator PORV or safety valve, (2) main steamline break, (3) feedwater line break, (4) inadvertent operation of the ECCS, and (5) steam generator tube rupture. Based on its evaluation of analytical results for the depressurization transients in the FSAR at selected power levels, the licensee found that the impact of operation of SI pumps on the results of the transient is insignificant even for full power operation. Therefore, the licensee concluded, and the staff agreed, that use of one SI pump does not change the conclusions for the current FSAR transient analysis, which assumed two SI pumps available for SI flow.

2.1 <u>Large Break Loss-of-Coolant-Accident Analysis Evaluation</u>

The licensee provided the results of a large break LOCA analysis supporting the request for operation up to 1380 MWt (60% power).

The licensee analyzed and evaluated the double ended cold leg guillotine (DECLG) break with a discharge coefficient of 0.4, since this break was identified previously as the limiting case resulting in the highest peak cladding temperature (PCT). The DECLG break analysis was performed with a power peaking factor (F_Q) of 2.26, 102% of the full power of 2300 MWt, and an assumed loss of offsite power at the beginning of the accident. To satisfy the worst single failure criteria, the licensee assumed only one SI pump available in the analysis. The flow changes resulting from the use of one SI pump rather than two have no affect on the limiting break pipe. The analysis was performed by using the modified version of the 1981 Westinghouse ECCS evaluation model with inclusion of the BART methods, which were previously approved by NRC.

The staff has reviewed the large break LOCA analysis and found that (1) the calculated PCT is 2198.5° F which is less than the acceptance criteria of 2200° F, (2) the maximum local metal-water reaction is 7.14 percent which is below the limit of 17 percent, and (3) total core metal-water reaction is less than 0.3 percent which does not exceed the acceptable limit of 1.0 percent. Since approved methods and computer codes were used and the analytical results are within the acceptance criteria of 10 CFR 50.46, this analysis would demonstrate that for a large break LOCA the plant can still satisfy

10 CFR 50.46 at full power, with some adjustment of the peaking factor, with one SI pump (two pumps but assuming a single failure). The case at 100% power bounds the case at 60% power because of the lower fuel temperature, lower decay heat and lower stored energy in the core at 60% power, and the large break LOCA analysis presented for 2300 MWt is adequate to support plant operation at 1380 MWt (60% power).

2.2 Small Break Loss-of-Coolant-Accident Analysis Evaluation

The limiting accident for the revised SI configuration is the small break LOCA. Initially, there had been some indication that the plant might still satisfy 10 CFR 50.46 at or near full power relying on manual actuation of the swing SI pump. However, after further review the licensee subsequently proposed the changes discussed below including a restriction in power to 1380 MWt.

The small break LOCA (SBLOCA) analysis was performed with the approved codes; i.e., (1) NOTRUMP for the calculation of the transient depressurization of RCS, core power, water-steam mixture height and steam flow past the uncovered portion of the core, and (2) LOCTA for the PCT analysis. Three small break LOCA analyses were done assuming 102% of 1380 MWt (60% of full power) and assuming one HHSI pump available for delivery of the SI flow. These analyses were performed for 2.0-inch, 1.5-inch and 1.0-inch equivalent diameter breaks. The 2.0-inch case had the highest PCT of 965.4° F.

An analysis was also performed for a 3.0-inch break, which was previously identified as the limiting SBLOCA case, with full power and only one SI pump available. The results for the 3.0 inch break at full power showed that the PCT is 1772° F, which is within the acceptance criteria of 2200° F. The licensee indicated, and the staff agreed, that the 3.0-inch case at 100% power will bound the 3.0-inch case at 60% power.

The staff has concluded that the small break LOCA analyses are acceptable since the approved method was used, a sufficient break spectrum was analyzed, and the analytical results for all cases of operation at 1380 MWt are within the acceptance criteria of 10 CFR 50.46.

2.3 Technical Specifications Changes

The evaluation of the Technical Specifications changes submitted follows:

(1) Technical Specification 3.3.1.1.c

The licensee proposed adding a note to this section, which restricts the operating power up to 1380 MWt for the conditions with only two SI pumps operable (each capable of automatic initiation from a separate emergency bus). This change is acceptable since the change is supported by the acceptable analysis discussed in this evaluation. In addition, the note states that, prior to exceeding 1380 MWt, NRC review and approval is required. This condition is acceptable to assure adequate emergency core cooling capability to support operation above 1380 MWt.

(2) Technical Specifications 3.3.1.2.b

Additional surveillance requirements and corrective action are provided for operation up to 1380 MWt assuming the loss of one of the two SI pumps which is required to be operable. The changes are consistent with the current Technical Specifications required for loss of an SI pump and are acceptable.

(3) Technical Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 and Figure 3.10-3.

Notes are added to the related sections regarding reduction of the power peaking factor (F_Q) from 2.32 to 2.26 for conditions with only two SI pumps operable above 50% of full power. The corresponding value for power levels less than 50% of full power decreases from 4.64 to 4.52. The corresponding Axial Power Distribution Monitoring System value is reduced proportionally from 2.103 to 2.049. These changes are consistent with the assumptions used in the supporting analysis and are acceptable.

3.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether or not a no significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from an accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The following evaluation in relation to the three standards demonstrates that the proposed amendment does not involve a significant hazards consideration.

1. The proposed amendment would not involve a significant increase in the probability or consequences of any accident previously evaluated. Prolonged operation at 1380 MWt would not result in significant changes in the flow conditions of the reactor coolant system that could increase the probability of an accident. The H. B. Robinson plant had an extensive history of reduced power operation prior to the steam generator replacement in 1984 and experienced no condition that could increase the probability of an accident.

As indicated above, the changes affect large break and small break LOCA accident sequences and have little or no affect on other accidents and transients. The proposed changes do not increase the probability of either a large or small break LOCA. The SI system is a part of the emergency core cooling systems designed to mitigate the effects of loss of coolant accidents in the event such accidents should occur. The changed configuration does not affect the probability of pipe rupture or any other initiating event leading to a loss of coolant. The power restriction compensates for the elimination of automatic transfer of the swing SI pump and assures that plant operation will satisfy emergency core cooling system requirements with adequate reliability to satisfy the single failure requirements of 10 CFR 50.46 and Appendix K and of 10 CFR Part 50 Appendix A General Design Criterion 35.

With power limited to 1380 MWt, in the event of either a large break or a small break LOCA, the calculated peak clad temperature will be within the limits of 10 CFR 50.46(b), using widely used calculational methods which have been previously approved by the NRC staff. This will assure that the consequences of the only accidents affected by the changes will not significantly increase over those previously analyzed in the Analysis Section of the H.B. Robinson Final Safety Analysis Report (UFSAR). In fact, the power restriction should serve to reduce calculated consequences somewhat.

2. The SI system is a part of the emergency core cooling systems designed to mitigate the effects of loss of coolant accidents in the event such accidents should occur. The changed configuration does not affect the probability of pipe rupture or any other initiating event leading to a loss of coolant. Its only affect is on SI system response to previously analyzed accident sequences and as discussed above, with the compensating power restriction, the changes involved in this amendment do not significantly increase the probability or consequences of such previously analyzed accidents.

3. Operation of the facility, in accordance with the proposed amendment, would not involve a significant reduction in a margin of safety. The analysis of reduced power operation has shown that postulated failures will not produce plant conditions which exceed the safety parameters specified in the Accident Analysis of the UFSAR. Specifically, there is no significant reduction in safety margin on the reactor core parameters, such as peak fuel clad temperatures, during postulated accidents for the proposed amendment in comparison with those prior to the amendment for full power operation with three operable SI pumps.

Based on the foregoing, the Commission has concluded that the standards of 10 CFR 50.92 are satisfied. Therefore, the Commission has made a final determination that the proposed amendment does not involve a significant hazards consideration.

4.0 FINDING ON EXISTENCE OF EMERGENCY SITUATION

The regulations at 10 CFR 50.91(a)(5) provide the necessary requirements for issuing an amendment when the Commission finds that an emergency situation exists and failure to act in a timely way would result in derating or shutdown of a nuclear plant. The Commission expects its licensees to: apply for license amendments in a timely fashion; not abuse the emergency provisions by failing to make a timely application for the amendment and thus itself creating the emergency; and provide an explanation as to why the emergency situation occurred and why it could not have been avoided.

The H. B. Robinson plant has been shutdown since January 29, 1988, when the licensee identified that the SI system did not meet the single failure criterion to support full power operation. This previously unanalyzed condition was identified during the licensee's review of the SI system control logic in response to an NRC request for information related to the emergency electrical distribution to the SI pumps. The NRC request was made on January 14, 1988.

The basic design for emergency electrical distribution to the SI system has not been changed since the plant was licensed to operate. Prior to the review, the licensee had no knowledge that the single failure criterion could not be met for full power operation. Once the problem was identified and the plant placed in cold shutdown, the licensee promptly evaluated and made several modifications to the emergency distribution system control logic which would restore the SI system for full power operation under a number of single failure scenarios. However, following the completion of modifications on February 16, 1988, there remains one scenario for which the licensee has not identified a method for resolution. The licensee also informed the NRC staff that near-term resolution to this remaining single failure scenario is not expected. In order for the plant to restart with only two operable SI pumps, evaluation shows that the power level has to be restricted to no more than 1380 MWt. Consequently, by letters dated February 24, February 26, 1988 and March 1, 1988, the licensee requested that an emergency amendment to place restrictions on power level be processed to allow the plant to restart when only two SI pumps are operable.

Unrelated to the SI system, the licensee has experienced over-speed trips of the emergency diesel generators (EDGs) during fast start tests. The plant will not be restarted until the EDGs have been repaired and their operability verified. The licensee's schedule for plant heatup is on or about March 4, 1988. Therefore, an emergency license amendment is required to avoid delay of startup of the plant.

The staff has reviewed the licensee's explanation of the circumstances justifying consideration of this amendment on an emergency basis. Based on this review, the staff finds that the licensee used its best efforts to apply for the subject amendment in a timely manner and that it had not acted in a manner to create the emergency to take advantage of these procedures.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changed a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site; and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) these amendments will not (a) significantly increase the probability or consequences of accidents previously evaluated, (b) create the possibility of a new or different accident from any previously evaluated, or (c) significantly reduce a margin of safety and, therefore; the amendments do not involve significant hazards considerations; (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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

The staff consulted with the State of South Carolina and the State of South Carolina did not have any comments.

Principal Contributors:

Robert Jones
Summer Sun
Peter Kang

Dated: March 7, 1988

AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-23 H. B. ROBINSON, UNIT 2

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