Docket

DEC 3 1975

DISTRIBUTION

Docket
NRC PDR
LOCAL PDR
ORB READING
OELD
OI&E (3)
NDUBE
BJONES (4)
JMcGOUGH
JSALTZMAN, OAI (w/o Tech Specs)
RINGRAM
RWREID
KRGOLLER
SKARI(w/o Tech Specx)
BSCHARF(15)
TJCARTER
PCOLLINS
SVARGA
CHEBRON
AESTEEN
DEISENHUT
GRAY FILE
EXTRAS (5)
ABERNATHY
JRBUCHANAN
DBRIDGES
MDUNCAN

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 15 to Facility License No. DPR-23 for the H. B. Robinson Unit 2 facility. This amendment includes Change No. 40 to the Technical Specifications, and is in response to your request dated October 14, 1975 as supplemented by correspondence dated August 3, August 22, October 17, November 13, November 18, and November 24, 1975.

This amendment establishes operating limits for Cycle 4 operation in the Technical Specifications based upon an acceptable evaluation model that conforms to the requirements of 10 CFR 50.46, and revises provisions related to the replacement of 52 fuel assemblies in the Robinson-2 core with fuel assemblies of a different design, constituting refueling of the core for operation with Cycle 4.

We have evaluated the potential for environmental impact of plant operation in accordance with the enclosed amendment, and 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 pursuant to 10 CFR Part 51, 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

Robert W. Reid, Chief
Operating Reactors Branch 4
Division of Reactor Licensing

Enclosures:
1. Amendment No. 15
2. Negative Declaration
3. ~~Environmental Impact Appraisal~~
4. Safety Evaluation
5. Federal Register Notice

cc: see next page

Conc't

	ORB 4	TR/RSB		AD/RL
	RIngram	ZRR		KGoller
	11/26/75	12-1		12/3/75

OFFICE	ORB 4	TR/CPB	OELD	ORB 4
SURNAME	DBRIDGES/BK	for DFK	[illegible]	RWReid
DATE	11/28/75	11/28/75	12/2/75	12/3/75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc w/enclosures:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
Barr Building
910 17 Street, N. W.
Washington, D. C. 20006

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29532

Hartsville Memorial Library
Home and Fifth Avenues
Hartsville, South Carolina 29550

cc w/ enclosures and incoming:
Dated October 17, 1975, November 13, 1975
November 18, 1975 and November 24, 1975

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
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 October 14, 1975, as supported by correspondence dated August 3, August 22, October 17, November 13, November 18, and November 24, 1975, 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; and

   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.

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

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

" B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 40 ."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 40 to the
Technical Specifications

Date of Issuance:

December 3, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 15

CHANGE NO. 40 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.3-2	2.3-2
3.1-15	3.1-15
3.1-17	3.1-17
3.1-22	---
3.3-3	3.3-3
3.10-1	3.10-1
3.10-9 - 3.10-14	3.10-9 - 3.10-14
3.10-17	3.10-17

(d) Overtemperature ΔT

$$\leq \Delta T_o \{K_1 - K_2 (T - 574.0) + K_3 (P - 2235) - f(\Delta I)\}$$ 40

where:

ΔT_o = Indicated T at rated power, ^oF

T = Average temperature, ^oF

P = Pressurizer pressure, psig

K_1 = 1.095

K_2 = 0.0107 40

K_3 = 0.000453

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion ~~chambers;~~ with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $(q_t - q_b)$ within +12% and -17% where q_t and q_b are percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power; $f(\Delta I) = 0$. For every 2.4% below rated power level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power level; the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power.

(e) Overpower ΔT

$$\leq \Delta T_o \{K_4 - K_5 \frac{dT}{dt} - K_6 (T - T') - f(\Delta I)\}$$

where

ΔT_o = Indicated ΔT at rated power, ^oF

T = Average temperature, ^oF

3.1.5 LEAKAGE

Specification:

3.1.5.1 If the primary system leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.2 If the sources of leakage have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.1.5.3 If the primary to secondary leakage in a steam generator exceeds 1 gpm, the reactor shall be placed in the hot 40
shutdown condition within 8 hours utilizing normal operating procedures. If the leakage exceeds this limit for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Basis:

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System, and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR Part 20 would be exceeded. This is shown as follows: If the reactor coolant activity is $50/\bar{E}$ uCi/cc ($\bar{E}$ = average beta plus gama energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.00 \times 10^{-5}$ sec/m^3 is about the 10 CFR Part 20 guideline of 0.5 R/yr[1,2].

With the limiting reactor coolant activity and assuming initiation of 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks, from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 0.5 gpm to greater than 10 gpm can be detected and measured by this system. Condensate flow corresponding to coolant leakage of approximately 1 gpm can be detected within 10 minutes.

Leaks less than 1 gpm can be measured by periodic observation of the level changed in the condensate collection system.

If leakage is to another closed system, it will be detected by the plant radiation monitors and/or inventory control.

40

Operator action to start to place the reactor in the hot shutdown condition within 12 hours utilizing normal operating procedures provides adequate time for an orderly reduction of power. The hot shutdown condition allows personnel to enter the containment and inspect the pressure boundary for leaks. The 24 hours allowed prior to the operator starting to place the reactor in the cold shutdown condition utilizing normal operating procedures allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown condition would be in order.

a. One accumulator may be isolated for a period not to exceed 4 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.

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. If the boron concentration in the boron injection tank falls below 20,000 ppm, and is greater than 15,000 ppm, the reactor may remain in operation for a period not to exceed 24 hours. If the concentration is less than 15,000 ppm, the reactor will
be placed in the cold shutdown condition utilizing normal 40
operating procedures.

g. Power or air supply may be restored to any valve referenced in 3.3.1.1.h. and 3.3.1.1.i. 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 4 hours.

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures.

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specifications:

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be no further inserted than the limits shown by the solid lines on Figure 3.10-1 for 3 loop operation.

3.10.1.4 At 50% of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core life values as shown by the dotted lines on Figure 3.10-1. 40

3.10.1.5 Except for physics tests, if a part-length or full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

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% of rated power for 3 loop operation.

3.10.1.6 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained except for 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 and part length rods fully withdrawn.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of hypothetical rod ejection and provide for acceptable nuclear peaking factors. The solid lines shown in Figure 3.10-1 meet the shutdown
requirement for the first 50% of the cycle. The end-of-cycle life 40
limit is represented by the dotted lines. The end-of-cycle life limit may be determined on the basis of plant startup and operating data to provide a more realistic limit which will allow for more flexibility in plant operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the
shutdown margin required at end of life with respect to an un- 40
controlled cooldown. All other accident analyses are based on 1% reactivity shutdown margin. The specified control rod insertion
limits meet the design basis criteria on (1) potential ejected 40
control rod worth and peaking factor[4], (2) radial power peaking factors, $F_{\Delta H}$, and (3) required shutdown margin.

The various control rod banks (shutdown banks, control banks, and part length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position[2]. The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions. 40

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.1.6) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of linear power density must not exceed 21.1 kW/ft. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the ECCS Acceptance Criteria. To aid in specifying the limits on power distribution the following hot channel factors are defined.

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^{N}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^{N}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes through the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^{N}$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided the values of F_q and $F_{\Delta H}$ in Specification 3.10.2.1 are not exceeded. 4

For normal operation, it is not necessary to measure these quantities. Instead, it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as shown in Figure 3.10-1.

3. The control bank insertion limits are not violated.

4. Part length control rods are not inserted.

5. Axial power distribution control procedures, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined on the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power, the design limits are met, provided the limits of Specification 3.10.2.1 are not exceeded. 4

The permitted relaxation in $F_{\Delta H}^{N}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset, it varies primarily with burnup.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with control Bank D more than 190 steps withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the specified deviation of ΔI 40
is permitted from the indicated reference value. During periods where extensive load following is required, it may be impossible to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power.

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled with the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents. 4

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

An upper bound evelope of peaking factors has been determined from extensive analysis considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10.2. The specifications on power distribution control insure that xenon distributions are not developed which, at a later time could cause greater local power peaking even though the flux difference is then within limits. The results of a loss of coolant accident analysis based on this upper bound envelope indicate that a peak clad temperature would not exceed the 2200°F. limit. The nuclear analyses of credible power shapes consistent with the power distribution control procedures have shown that the F_q^T limit is not exceeded. 4 40

For transient events the core is protected from exceeding 21.1 KW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

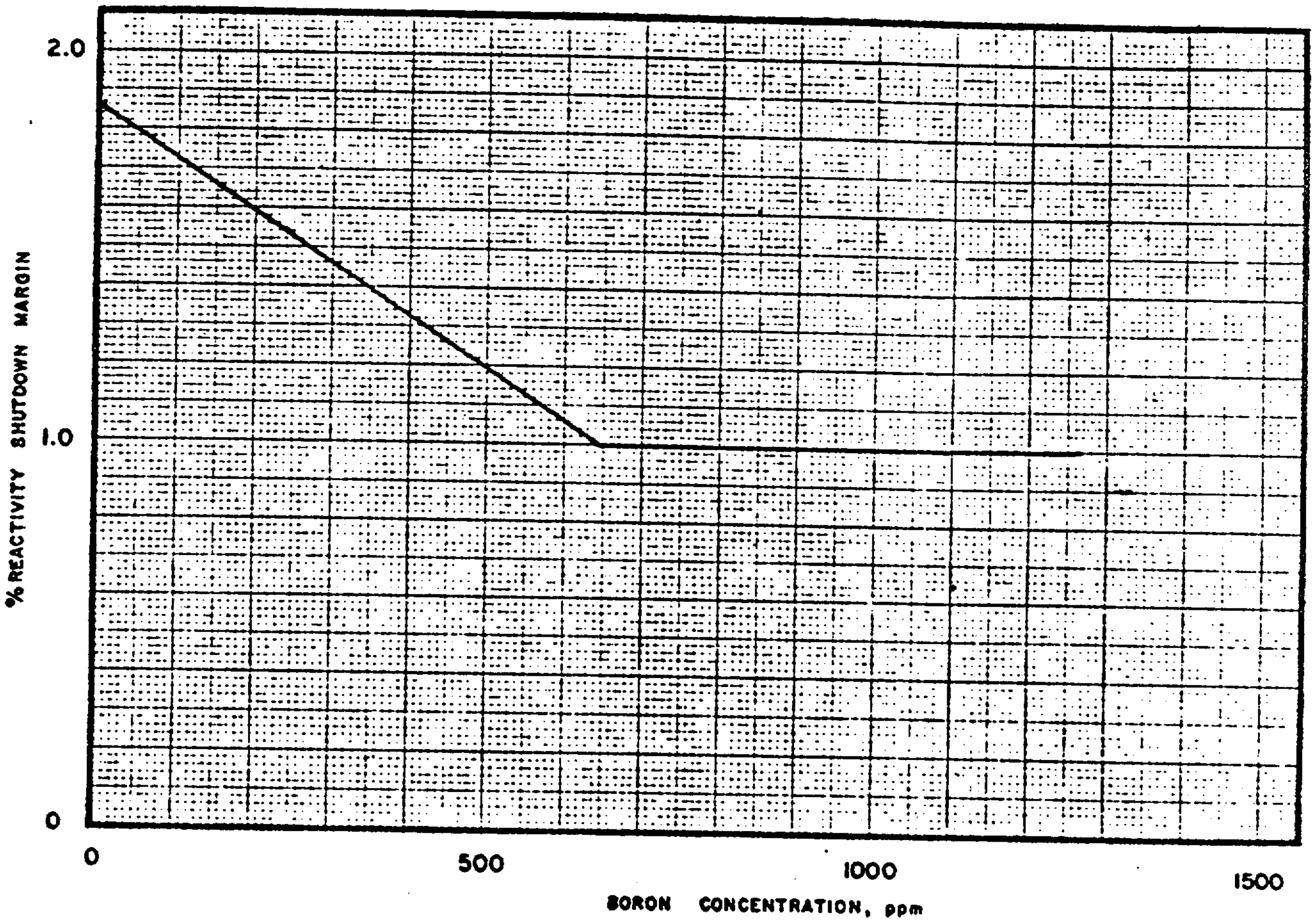
In the specified limit of F_q^N there is a 5% allowance for uncertainties(1) which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured F_q^N 5% less than the limit for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for and 5% is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system. 40

In the specified limit of $F_{\Delta H}^{N}$ there is an 8% allowance for design prediction
uncertainties which means that normal operation of the core is expected to
result in $F_{\Delta H}^{N}$ at least 8% less than the limit at rated power. The uncertainty 4
to be associated with a measurement of $F_{\Delta H}^{N}$ by the movable incore system on the
other hand is 4% which means that the normal operation of the core shall result
in a measured $F_{\Delta H}^{N}$ at least 4% less than the value at rated power. The logic 40
behind the larger design uncertainty in this case is that (a) abnormal
perturbation in the radial power shape (e.g., rod misalignment) affect $F_{\Delta H}^{N}$
in most cases without necessarily affecting F_q^N through movement of part
length rods and can limit it to the desired value (b) while the operator has
some control over F_q^N through F_z^N by motion of control rods, he has no direct
control over $F_{\Delta H}^{N}$, and (c) an error in the predictions for radial power shape
which may be detected during startup physics tests can be compensated for in F_q^N
by tighter axial control, but compensation for $F_{\Delta H}^{N}$ is less readily available.

Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions, measured as part of the startup physics testing, are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

During normal plant startup, quadrant power tilt ratio may exceed 1.02 due to instrumentation instabilities as a result of rodded configurations and low excore detector signal levels below 50% of full power. Sustained power operation below 50% of full power would require a renormalization of the calculational methods for determining power tilt to compensate for change in signal levels once equilibrium conditions are met.

The two-hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt conditions cannot be eliminated within the two-hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system. For a tilt condition $\leq$ 1.09 an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.



40

Figure 3.10-2 SHUTDOWN MARGIN vs BORON CONCENTRATION

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-23

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

DOCKET NO. 50-261

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-23. These changes would authorize the Carolina Power and Light Company (the licensee) to operate the H. B. Robinson Steam Electric Plant, Unit 2 (located in Darlington County, Hartsville, South Carolina), with limiting conditions associated with fuel assembly specific power resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS).

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-23, H. B. Robinson Steam Electric Plant, Unit 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for H. B. Robinson Steam Electric Plant, Unit 2, published in April 1975.

The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina.

Dated at Rockville, Maryland, this 2nd day of December 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

William H. Regan, Jr., Chief
Environmental Projects Branch #4
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 15 TO DPR-23

CHANGE NO. 40 TO THE TECHNICAL SPECIFICATIONS

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated October 14, 1975 and March 14, 1975, Carolina Power and Light Company (CP&L) submitted proposed changes to the Technical Specifications, Appendix A, to License DPR-23. The proposed changes resulted from operation with Cycle 4 fuel which will include both new Exxon Nuclear Company (ENC) and recycled Westinghouse Company fuel assemblies. Supplemental information relating to the reload and the Emergency Core Cooling System (ECCS) analysis have been provided by CP&L in their correspondence dated September 24, 1974, August 3, August 22, October 17, November 13, November 18, and November 24, 1975. The ECCS analysis with the ENC model for the Cycle 4 core indicates that there are no limit changes from the earlier approved Westinghouse ECCS analysis of October 17, 1975. The licensee states that there is no environmental impact associated with this ENC fuel reload. We have independently reviewed this matter and the conclusions are set forth below.

CP&L is presently licensed to operate H. B. Robinson Steam Electric Plant, Unit 2, located in the State of South Carolina, Darlington County, at power levels up to 2,200 megawatt thermal (MWT). Operation with the proposed ENC reload core does not result in an increase or decrease in power levels of the unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statement (FES) for H. B. Robinson Steam Electric Plant, Unit 2, Docket No. 50-261, dated April 1975.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with the operation of the Cycle 4 core within ECCS acceptance criteria as calculated with the ENC ECCS calculational model. No ECCS limits are changed as a result of calculation with the ENC model.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statement (FES) for H. B. Robinson Steam Electric Plant, Unit 2, Docket No. 50-261, dated April 1975, can be predicted for the proposed action. The staff's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected radionuclide production and their release rates to the environment. The estimates of radionuclide production and their release rates are not significantly affected as the licensed reactor power is unchanged. No increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for H. B. Robinson Steam Electric Plant, Unit 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO LICENSE NO. DPR-23

CHANGE NO. 40 TO TECHNICAL SPECIFICATIONS

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

INTRODUCTION

By letter dated October 14, 1975, Carolina Power and Light Company (CP&L) proposed changes to the Technical Specifications appended to Facility Operating License DPR-23 for H. B. Robinson Steam Electric Plant Unit No. 2 (Robinson 2). Supplemental information relating to the requested changes was supplied by CP & L in their letters of September 24, 1974, August 3, August 22, October 17, November 13, November 18, and November 24, 1975. The purpose of the requested changes is to revise the Robinson-2 Technical Specifications to permit operation during the fourth fuel cycle (Cycle 4) with new reload Exxon Nuclear Company (ENC) fuel assemblies and recycled Westinghouse Company fuel assemblies.

DISCUSSION

The Robinson-2 reactor core consists of 157 assemblies, each having a 15x15 array of fuel rods. The Cycle 4 reload will consist of 105 Westinghouse assemblies and 52 Exxon Nuclear Company assemblies. The Westinghouse assemblies will include a two-cycle exposure (Region 4) assembly located in the core center position, and two one-cycle exposure regions (Regions 5 and 6 - 52 assemblies each) scatter-loaded throughout the core interior. The 52 fresh ENC assemblies (Region 7) will be located at the core periphery.

Technical information has been provided by CP & L which includes a general description of the reload core, detailed mechanical design data on the reload fuel, nuclear and thermal-hydraulic data, accident and transient analyses, and the loss of coolant accident analysis in support of the reload application. Since this is the first reload application of ENC fuel to PWR's ENC has provided documentation[1-9] of the ENC nuclear design methods, the loss of coolant accident analysis models and the computer codes employed for the analyses.

We have examined the methods employed by ENC and conclude that their application to the design and analyses of the H. B. Robinson Unit 2 Cycle 4 reload is satisfactory. Further, following our review of the available reload information we conclude that it is acceptable for the licensee to proceed with Cycle 4 operation. Our review and evaluation is discussed in the following paragraphs.

EVALUATION

1. Reactor Core Description

The fuel to be added to the core is not significantly different in design or in operating characteristics from the original fuel it replaces. CP & L's analysis of the of the loading pattern and their analysis of the core physics parameters indicate that the nuclear parameters for Cycle 4 fall within the range of values assumed in the Robinson-2 Final Safety Analysis Report (FSAR). We have reviewed the calculations for the proposed loading pattern, calculational techniques for computing various core parameters, comparison of calculated values to measured values for many of these parameters, and conclude that the proposed operating limits and values for the reactor core parameters used in the transient analyses are adequately conservative and are acceptable.

2. Fuel and Mechanical Design

The ENC 15x15 fuel assemblies like the Westinghouse assemblies have 204 fuel rods, 20 Zircaloy-4 guide tubes and one Zircaloy-4 instrumentation tube. A comparison of the mechanical designs for the Exxon (Region 7) and Westinghouse (Regions 5 and 6) assemblies indicates that the only noteworthy difference in the fuel rod design is the use of thicker cladding for the ENC fuel (30 mils for Exxon compared to 24.3 mils for the Westinghouse design) which is more conservative. The same dimensions in all critical areas have been maintained. The increased cladding thickness is accommodated primarily by a decreased pellet diameter and slightly larger cladding diameter (2 mils). The Exxon fuel assembly spacer grids are comprised of a Zircaloy-4 structural frame with Inconel springs and constitute a five point rod support system. The Westinghouse design is all Inconel and has a six point support system. Exxon has evaluated the fretting wear performance of the spacers and fuel rods in a flow test operated at maximum reactor conditions for more than 1,000 hours. No signs of fretting corrosion were observed in the fuel rods inspected.

Exxon has also performed hydraulic flow tests to evaluate the compatibility between Westinghouse Region 6 and Exxon Region 7 fuel assemblies.[10] The results of these tests show that although there were some differences in the pressure drop distributions between the upper and lower tie plates and the bare rods and spacers the flow through the Exxon assembly is within 3% of that in the Westinghouse Region 6 assemblies. This difference of flow has been considered in the analysis and this flow differential is acceptable.

Fuel performance calculations that account for the effects of fuel densification (namely the potential for cladding collapse into axial gaps, the increase in the linear heat generation rate, the increase in the stored energy, and the increased probability of local power spike resulting from axial gaps) have been performed with the NRC staff approved version[11] of the Exxon Nuclear Company densification report.[12] The primary effects of densification on the fuel rod mechanical design are manifested in calculations of fuel-clad gap conductance and cladding collapse time. The calculation of gap conductance by the approved analytical model incorporates time-dependent fuel densification, gap closure, and cladding creepdown. The approved Exxon Clad Collapse Model[12] was used to calculate the collapse times for the Exxon fuel rod design. The cladding collapse time was calculated assuming the statistically worst clad geometry that can occur, the minimum initial fill gas pressure and taking no credit for fission gas release, and was determined to be in excess of maximum life exposure of the fuel rod.

H. B. Robinson Unit 2 is one of the first plants to use Exxon PWR type reload fuel assemblies. To date, other operating experience with Exxon PWR fuel has been two lead assemblies in the Ginna reactor. These assemblies were inspected after one cycle and had no leakers. CP & L plans a surveillance program for the Exxon fuel such that representative assemblies at Robinson will be inspected during future refueling outages to verify that the fuel is operating satisfactorily. We concur that the surveillance program by CP & L will provide adequate information on Exxon fuel performance.

We have compared the features of the Westinghouse and Exxon fuel assembly designs for Robinson to determine how they specifically relate to rod bow. Both designs have the same number of spacers and unsupported span between the spacers. Design differences are listed in the reload submittal and these features which Exxon believes will reduce the extent of creep bow, include:

(1) Thicker cladding
(2) Slightly larger diameter
(3) Deeper grid spacers (larger rotational restraint)
(4) Five point support system compared to a six point system used by Westinghouse (reduced axial restraint).

We are in agreement with Exxon that the thicker cladding and larger rod diameter should contribute to the reduction of rod bow. However, with regard to the use of deeper spacer grids and fewer rod spacer contact points in the Exxon design we have concluded that the current understanding of rod bow is insufficient to assess the effects of these design differences.

Rod bow data available to the NRC staff from operating reactors has shown a definite time dependency. Irradiation of the Westinghouse assemblies will precede that of the Exxon reload assemblies by one or more cycles.

Due to this time dependency, the similarities in the Exxon and Westinghouse designs and the thicker Exxon cladding, it is our opinion that the new Exxon fuel assemblies will bow less than any of the Westinghouse assemblies during Cycle 4 operation.

Based on the review of the Cycle 4 reload submittal and supplemental information, we conclude:

1. The Exxon fuel rod mechanical design is compatible with the previously approved Westinghouse fuel design and provides acceptable engineering safety margins for normal operation;

2. The effects of fuel densification have been acceptably accounted for in the fuel design;

3. The rod bow in the Exxon fuel assemblies during the first cycle of operation should be less than for the Westinghouse fuel assemblies all of which have been in reactor one or more cycles.

We approve the Cycle 4 reload of Robinson-2 with the previously approved Westinghouse fuel assemblies in Regions 4, 5 and 6 and with the new Exxon fuel assemblies in Region 7 as stated in the reload submittal. The rod bowing phenomenon will be more predominant in the exposed Westinghouse fuel and, hence, that fuel will be controlling in that regard. This matter is addressed in a later section in this evaluation.

3. Nuclear Analysis

The methods employed by Exxon Nuclear Company for the neutronic design of H. B. Robinson Cycle 4 reload are described in references 1-3. We have reviewed these methods and conclude that their application of Cycle 4 is acceptable. These methods enable the nominal values and range of the neutronic characteristics to be predicted with sufficient accuracy to permit selection of conservative values to be used for accident analyses.

The licensee chose to use Cycle 2 as a reference for neutronic characteristics because Cycle 3 contained two regions which did not reflect a typical reload situation. The predicted characteristics for Cycle 4 are very similar to those throughout Cycle 2. CP & L has proposed to verify the accuracy of Beginning of Cycle (BOC) predictions during startup tests.

Azimuthal plane peaking factors, F_{xy}, were predicted for various operating states for Cycle 4. The largest value expected for this parameter during operating conditions was 1.384. This meets the loading pattern design criterion to keep the maximum unrodded F_{xy} less than or equal to 1.425 and will be verified during startup measurements. In determing the total peaking factor, F_Q, a value of 1.435 was used for the unrodded plane radial peaking factor, F_{xy}.

The F_Q used to determine the maximum initial linear power density for the LOCA analysis was 2.30, as in the latter part of Cycle 3. Control and monitoring of the power distribution will continue to be accomplished using constant axial offset control (CAOC) as in the latter part of Cycle 3. No changes in the Technical Specifications for CAOC are required for Cycle 4. The F_Q of 2.30 is slightly less than the value of 2.32 accepted for the generic application[13] of CAOC. The lower value is acceptable because the analysis of CAOC for operation without part length rods (as specified in the Robinson Technical Specifications) in a depleted core shows considerable margin to the 2.32 envelope. With no part length rods the 2.32 envelope is approached only for BOC first cycle cases.

The control requirement and control rod worth analysis furnished by the licensee indicates margin between the most limiting shutdown requirement and predicted control rod worth. The maximum shutdown requirement is 1.83% Δk for the credible steam line break accident at the end of the cycle. The margin between the total rod worth assuming that the most reactive control rod is stuck, and the total reactivity requirement, is at least 2.61% Δk at the end of the cycle. In addition, the prediction of the total control rod worth is conservative because it contains a 10% allowance for calculational uncertainty. Startup measurements of some of the control rod banks will be made to confirm that the predicted control rod worth is realized for Cycle 4. Therefore, we conclude the shutdown margin requirements will be met in Cycle 4 operation, and the proposed Technical Specification change to reflect allowance of 1.83% Δk shutdown margin is acceptable.

4. Thermal Hydraulic Analysis

The thermal-hydraulic analysis by CP & L indicated the following results:

a. The ENC and Westinghouse fuel are thermally and hydraulically compatible.

b. The minimum departure from nucleate boiling ratios (MDNBR) for both fuel types are always greater than 1.30 for normal operation and anticipated transients.

The thermal-hydraulic analysis included both experimental measurements and theoretical calculations. We have reviewed the experimental setup, experimental results, and analytical evaluations and concur that the two fuel types are compatible. The interaction between the two fuel types is only slight and flow between and ENC fuel assembly and a Westinghouse Region 6 assembly is within 3%. This difference of flow has been considered in the analysis and this flow differential is acceptable.

The adequacy of the ENC fuel for meeting MDNBR's requirements has been verified with transient analyses. The results of the transient calculations are discussed later in this evaluation.

DNB calculations performed independently by both ENC and Westinghouse indicate that the MDNBR is greater than 1.3 for both type fuel assemblies operated under the conditions of Cycle 4. Additional margin of at least 6% is provided by the fact that the DNB calculations were performed for operation at 2300 MWt while Robinson-2 is to be licensed for only 2200 MWt for Cycle 4. This margin is discussed further in the following section. The adequacy of the Westinghouse fuel has been submitted[14] previously by the licensee and has been determined to be acceptable.[15]

We find the MDNBR values acceptable (>1.30). We also conclude from our review that the other existing thermal hydraulic limits are acceptable for both fuel types.

5. Transient and Accident Analysis

The reload of the H. B. Robinson Unit 2 reactor with ENC fuel results in core parameters which differ slightly from previous reloads. To demonstrate that the reload fuel meets plant Technical Specifications during abnormal occurrences, transient analyses including the most limiting cases were performed using the ENC PTS-PWR code.[4] The analyses [16, 17] were performed assuming the reactor parameters of an equilibrium ENC fueled core for operation at 2300 MWt. The licensee submitted transient analyses for the Westinghouse fuel applicable for Cycle 4 in previous submittals,[14] and these analyses have been accepted[15] by the NRC. These analyses are still valid and remain acceptable.

The Technical Specification safety limits to be satisfied in transient analyses are peak system pressure of 2735 psia and an MDNBR of 1.30. The ENC-analyzed transient which resulted in the largest increase in system pressure was the loss of load transient. The peak pressure reached during this transient was 2530 psia, which was below the plant Technical Specification limit.

For all transients, the DNBR ratios did not decrease below the 1.3 limit. We have reviewed the ENC transient calculations, and we conclude that they are acceptable for application to the Robinson Cycle 4 reload.

6. ECCS Analysis

The present Cycle 3 Robinson-2 core is operating within limits[18] based upon an acceptable Emergency Core Cooling System (ECCS) evaluation model that conforms to the requirements of 10 CFR Part 50.46. However, since the proposed reload involves the use of ENC fuel an ECCS performance evaluation was conducted for the Cycle 4 reload core with an approved ENC ECCS evaluation model that meets the requirements of 10 CFR Part 50.46. The basis for acceptance of the ENC model are set forth in the NRC Safety Evaluation Report[19] on the ENC PWR ECCS Evaluation Model dated September 11, 1975 and the supplementary information and evaluation provided in Appendix A to this report. It is our determination that the Westinghouse and ENC fuel assemblies are sufficiently compatible such that the presence of the Westinghouse fuel in the Robinson-2 core does not impact on the ENC ECCS analysis. Likewise, the presence of the ENC fuel does not affect the results of the previous approved Westinghouse ECCS evaluation. The Robinson-2 evaluation conforms to the accepted model and meets the criteria established in 10 CFR Part 50.46 for peak clad temperature, maximum oxidation, and maximum hydrogen generation criteria.

The worst break location and single failure for Robinson-2 have been previously determined by Westinghouse sensitivity studies to be a pump discharge line severance with failure of one RHR pump and loss of offsite power. ENC has performed a series of break size calculations[20] at that location and assuming the same single failure. Calculations were performed for double ended guillotine breaks with discharge coefficients of 1.0, 0.6 and 0.4; for split breaks with areas of 8.24 ft^2 (equivalent in area to the double ended guillotine break of the pump discharge line), 3.0 ft^2, and 1.0 ft^2; and for small breaks with sizes of 1.0 ft^2, 0.5 ft^2, 8 inch diameter, and 6 inch diameter.

From the results of the above calculations, it has been determined that the 8.24 ft^2 split break is most limiting. We have reviewed the above results, and agree that the break spectrum has been defined sufficiently to assure that the worst break size for H. B. Robinson has been determined for the ENC code calculations, and we find the break spectrum calculations acceptable. The maximum peak clad temperature of 2066° F for the most limiting break is well within the limit of 2200°F presented in 10 CFR Part 50.46. Also the calculated maximum local metal-water reaction of less than 8% and total core metal-water of less than 1% are within the allowable limits of 17% and 1%, respectively. These calculations were done using a total peaking factor of 2.30.

Prior conclusions regarding the adequacy of the Robinson-2 ECCS which are unaffected by the use of ENC fuel remain valid. Therefore, it is our finding that operation with the reload core consisting of Westinghouse and ENC fuel assemblies is acceptable and fully meets the requirements of 10 CFR Part 50.46.

7. Rod Bow Penalty

The licensee has determined that a peak power penalty associated with fuel rod bowing must be assessed for the Westinghouse fuel assemblies. This penalty must be assessed since certain spacing or rearrangement of the fuel rods could cause power peaking on the order of 5% This penalty has been accounted for in two ways. First, because the ECCS performance evaluation was performed assuming the plant operates at 2300 MWt instead of the license limit of 2200 MWt, this difference in power can be traded for a 4.3% penalty. (An examination of the data showing dependence of bow penalty on burnup indicates that a maximum penalty of 4.3% should be assessed for assemblies irradiated up to approximately 23,000 MWD/MT). Secondly, the licensee has shown that the Robinson Cycle 4 fuel that will have EOC exposures greater than 23,000 MWD/MT will all operate at relative powers sufficiently below the maximum for the core to more than account for the additional 0.7% bow penalty predicted for exposures up to 33,000 MWD/MT.

In addition to the peak power penalty associated with the rod bowing phenomenon there has also been determined to be a DNB penalty resulting from the displaced coolant flow. This penalty is 5.6% maximum, but considering projected burnup of the highest burnup assembly this penalty is reduced to 2.8% for the Robinson-2 Cycle 4 Core. We have concluded that an additional 2% penalty should be imposed until the review of recently submitted Westinghouse rod bow information is completed. These penalties are easily accounted for by the above mentioned power limitation (100 MWt-4.3%) and by conservatisms in the present DNB model estimated to be on the order of 3.3%.

We find the manner in which the rod bow penalty has been assessed and accommodated acceptable.

8. Startup Tests

The startup tests will check the fuel loading and verify the calculational methods used in determining power distributions, shutdown margin and control rod worths.

Core flux maps at various power levels will be taken and evaluated to verify power distribution predictions. This data will also be used in establishing the excore/incore calibration.

The tests proposed to verify shutdown margin and control rod worths consist of determining the differential and integal rod worths for control banks D and C. (H. B. Robinson Unit 2 has control bank D, C, B and A and Shutdown banks A and B). In each of these tests a moveable detector map will be taken, the moderator temperature coefficient measured and the boron endpoint determined.

The licensee will also perform two additional tests to determine the worths of bank B and of the highest potential ejected rod worths from banks D, C and B.

The bank B determination will be similar to the tests for banks D and C. With this measurement the worth of 21 out of the 45 rods will be measured representing about 3.5% of the total 7.5% reactivity worth of all the rods.

The measurement of the highest worth ejected rod will verify the methodology for calculation of the worth of a single rod. Also the moveable detector map will be compared with the predicted distribution and the calculated tilt will be compared with the measured value.

The results from the startup tests will verify the calculational predictions and give a good indication of Cycle 4 performance. We find the proposed test program acceptable.

9. Miscellaneous Technical Specification Changes

Technical Specification changes have been proposed for a number of items that relate to the reload. Specific discussion follows:

a. Revision of overtemperature setpoint equation. The licensee proposes to modify the overtemperature ΔT reactor trip to correct a previous clerical error that inserted an equation that applied to operation at 2300 MWt (rather than the correct equation for the presently approved power level of 2200 MWt). The overtemperature ΔT is presently set by the equation $\Delta T \leq \Delta T_o(K_1 - K_2 (T - 575.4) + K_3 (P - 2235) - f(\Delta I))$ and is established to protect against Departure from Nucleate

Boiling. The licensee proposes to change the 575.4 value to 574 and to change the values for K_1, K_2, K_3. We have reviewed the proposed changes and concur that the proposed equation does reflect limits for operation at 2200 MWt and should be used rather than the present equation for 2300 MWt. This change will lower the ΔT set point on the order of a few degrees F and, hence, represents an increased conservatism in operation. We find the proposed change acceptable.

b. Revision of steam generator leakage limits. A Technical Specification change has been proposed that would delete the provisions for collapsed fuel in the core. All collapsed fuel was discharged from the Robinson-2 core at the end of Cycle 2. Subsequent reload fuel has been prepressurized to preclude collapse. The limits for steam generator leakage to minimize allowable leakage were in effect since the presence of collapsed fuel with certain levels of steam generator leakage could contribute to significant offsite dose levels for certain transients. We concur that the Technical Specification for the limits on collapsed fuel are not applicable to this reload core and are not necessary.

c. Revision of Technical Specifications for reactor shutdown when boron concentration in the boron injection tank (BIT) falls belows 15,000 ppm. Present Technical Specifications require that the reactor be placed in hot shutdown when the concentration in the BIT falls below 15,000 ppm. CP & L has proposed that the reactor be placed in cold shutdown instead of hot shutdown for this circumstance. This provides additional conservatism in their operation since going to cold shutdown would preclude the possibility of certain accidents and transients that could result with the reactor at hot shutdown with the reduced boron content in the BIT. We concur that the proposed measure is more conservative than their present limit and is acceptable.

d. Revision of Technical Specifications to make the Technical Specification wording independent of cycle. Wording changes have been proposed to delete reference to a specific reload cycle and, hence, make the Technical Specifications independent of cycle number. We have reviewed the proposed wording changes and concur that the wording changes have been limited to achieve that goal. This effort on the part of the licensee is an attempt to minimize Technical Specification changes for subsequent reload applications. We have reviewed the wording changes and concur that the proposed changes are acceptable.

SUMMARY

CP & L proposes to reload 52 new ENC assemblies and recycle 105 Westinghouse assemblies for the Robinson-2 Cycle 4 operation. We have reviewed the proposed loading pattern, the nuclear design calculations, methods, and analysis (including the loss of coolant accident analysis with an ENC approved model) and conclude that the proposed Technical Specification changes related to the reload are acceptable.

CONCLUSION

We have 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 or to the health and safety of the public.

Date:
December 3, 1975

I. Introduction

On September 11, 1975 the Nuclear Regulatory Commission issued a Safety Evaluation Report (1) presenting its review of the Exxon Nuclear Company's (ENC) generic PWR and H. B. Robinson Reactor ECCS evaluation models. The SER stated that some of the models were incomplete and/or inadequate and that the resolution of those items would be addressed in a Supplement.

The Exxon PWR Evaluation Model was considered by the full ACRS on September 12, 1975. The Committee letter (2) recommended approval of the proposed evaluation model providing the additional requirements set by the NRC Staff were satisfied. This Supplement discusses the resolution of these items for H. B. Robinson, and some further model modifications proposed since the date of the SER.

II. Evaluation Model Details

A) Momentum Equation Selections

The September 11, 1975 SER (1) noted in Sections 3.2.10 and 3.2.12 that ENC had chosen to specify the use of momentum equations which were not in conformance with the recommendations of "Evaluation of LOCA Hydrodynamics" (3). It was suggested that ENC perform its calculations using the recommended momentum equation selections, particularly in the junction connecting the upper downcomer and the broken cold leg pipe. On September 11, 1975, ENC submitted justification (4) for an alternative momentum equation scheme (5) which would simulate the 3-dimensional momentum flux vectors.

The ENC submittal stated that this treatment, while it preserves consistency in the use of complex volume flow area, also introduces the momentum flux contributions by the unidirectional volume associated with the junction in question. The submittal further stated that specifying the generalized momentum flux equation and a large flow area in the complex volume downcomer and plenums), while retaining the volume (cu. ft.) for the volume, establishes what may be considered a stagnation pressure for the volume and introduces only a small difference in frictional pressure drop.

Since pressure drop due to form loss contributions is the major contributor to the total pressure drop in the involved junctions, the staff accepted the ENC treatment. This conclusion was presented orally to the ACRS at its meeting on September 12, 1975. This equation form was used in the H. B. Robinson calculations and was used in the noding sensitivity study in the vicinity of the break.

B) Pool Boiling Logic Bypass and Dougall-Rohsenow 10% Limit for TOODEE2

In order to make the TOODEE2 small break blowdown heat transfer logic the same as RELAP4-EM, ENC was requested to insert a 10% lower quality limit on Dougall-Rohsenow and insert the same pool boiling logic bypass. When the pool boiling logic was bypassed, ENC determined that if the flow in TOODEE2 is identically zero the minimum heat transfer of 5 BTU/hr-ft^2 °F was also bypassed and a heat transfer coefficient of zero was used instead. Although, this is very conservative, it was suggested that ENC make the correction so the minimum heat transfer coefficient of 5 BTU/hr-ft^2°F would always apply during blowdown. Also, to assure that the effect of liquid in the upper plenum is not taken into account during downflow when a mixture level exists in the core, the minimum value is used above the mixture level in TOODEE2 at all times. These changes are acceptable.

ENC encountered some coding errors in TOODEE2 when the fuel-clad gap is closed during a transient calculation. Some of the corrections are documented in Appendix 5 and are correct. Other corrections were necessary and were made subsequent to Appendix 5. These changes also are satisfactory.

C). Model for Steam Temperature in the Steam Generator

Section 3.3.3 of the SER addresses a proposed numerical method for predicting average fluid temperature for each time step. This calculation was presented in the ENC REFLOOD model. This technique was reviewed and approved for application to the primary side fluid only. We have since been informed that the procedure was intended for use on the secondary side also. The staff has reviewed this extended application and concludes that the calculation is acceptable for both primary and secondary system fluid for the Robinson application.

D). Heat Slab Surface Temperature for Reflood

Section E of this Supplement discusses the ENC calculation for the refill period to bottom of core recovery (BOCREC). During this period, the detailed RELAP4-EM blowdown computation is discontinued. Beginning with the time of BOCREC, a RELAP4-EM FLOOD calculation commences which requires as input heat slab surface temperatures. Since these temperatures are not provided by the blowdown calculation, ENC sets these temperatures to the saturation temperature. Supplement 6 to XN-75-41 (6) documents a sensitivity study to show that this wall temperature initialization yields slightly conservative results when compared to temperatures obtained from a complete blowdown calculation. The uniform slab temperature results in slightly higher reflood rates than for the base case referenced in Appendix D of XN-75-41. The uniform slab temperature also results in slightly (≤20°F) lower cladding temperatures than the base case. However, the differences are not significant for Robinson and the staff finds the ENC treatment satisfactory for the Robinson submittal.

E). Time from End of Bypass (EOB) to Bottom of Core Recovery

ENC terminates its RELAP4-EM blowdown calculation at end of bypass, and performs a simplified calculation to evaluate the refill portion of the transient. ENC was requested to describe this computation subsequent to the September 11, 1975 SER. The end-of-bypass criteria, which uses zero velocity in the downcorner is predicted on the inspection of the blowdown sequence of events which generally shows end-of-bypass at about the time the the vessel is dried out. Any unusual behavior of blowdown (not evidenced in the Robinson case) showing unusually early end-of-bypass would warrant reexamination of the evidence.

The time to BOCREC computation is begun by determining ECCS flow rates using a 3-volume RELAP4 calculation. The input for this calculation includes accumulator inventory and system pressures at EOB from the blowdown calculation, and containment pressure from CONTEMPT-LT. With the ECCS flow rate and the EOB liquid inventory, together with system geometry data used for the blowdown analysis, the ECCS flow is integrated numerically to determine the time the liquid level is calculated to reach the bottom of the fuel. This time is amended to account for hot wall delay based on data presented in CREARE REPORT TN-188 (7), and considering cold leg spillage during hot wall delay. This latter spillage is calculated using system geometry configuration, and open channel flow methods (8).

The BOCREC time calculation is described in Supplements 5 (Revision 1) (9) and Supplement 7 (10). Supplement 6 (6) compares results from this method to those obtained from a RELAP4-EM extended blowdown run and shows that the simplified calculation is in good agreement and is slightly conservative. The methods have been reviewed by the staff and found acceptable.

F) Critical Flow Model During Reflood

Subsequent to the September 11, 1975 SER (1), ENC corrected a programming error in the RELAP4 Moody critical flow tables which resulted in unrealistic prediction of choked flow during reflood. Choking was predicted to occur at pressure ratios of about 1.23 which is considerably below the minimum value of 1.6 which is required for critical flow. ENC presented typical pressure ratios during reflood and concluded that critical flow should not occur. This correction, documented in Supplement 6 to XN-75-41 (6), has been reviewed by the staff and is acceptable.

G) Core Inlet Subcooling During Reflood

Supplement 6 to XN-75-41 (6) demonstrated that increased inlet ECCS subcooling at BOCREC result in conservative predictions of peak cladding temperature. The ENC method (11) of calculating ECCS inlet temperature considers heat addition from cold leg pipes is described in XN-75-41, Supplment 7, Section 6. Heat is also transferred to the water in lower plenum volumes after the liquid level has reached the volume midplane height. No credit is taken for heat added due to mixing with steam or residual water. In addition heat is not added in downcomer regions.

By the proposed ENC method, low amounts of heat are added to the ECCS water resulting in high inlet subcooling. Preliminary calculations show that the high subcooling results in lower core heat transfer coefficients which produce higher peak cladding temperatures. Trends in FLECHT data indicate that for some cases, the high inlet subcooling may result in lower peak clad temperatures. The staff finds the ENC calculation of ECCS inlet subcooling acceptable for the Robinson application but reserves judgment on the model for generic application in other plant applications until sufficient supporting data and/or calculations are presented.

H) Flooding Rate and Reflood Heat Transfer

Reflood rates and reflood heat transfer coefficients have been presented for the Robinson worst case break (1.0 double ended cold leg split). The flooding rate appears reasonable but is

50% higher prior to time of peak clad temperature than previously analyzed by Westinghouse. The heat transfer coefficient appears to be conservatively low, by comparison. The staff finds the overall calculations to be acceptable.

I) Core Flow Distribution

The September 11, 1975 SER (1) requires the review of input core flow parameters. A previous AEC contracted study (12) investigated RELAP4-EM PWR sensitivities, including core flow effects. With this as reference, the staff has reviewed core flow parameters used by ENC (13) for the Robinson analysis, and has found them acceptable.

III. Documentation

Subsequent to the September Safety Evaluation Report, ENC has provided Supplements 5 (revision 1), 6 and 7 (9, 6, 10) to describe and justify its PWR ECCS model. Supplement 7 presents a complete documentation of the model and shows the interrelation among the various computer programs. This documentation references several volumes in order to constitute the model.

IV. Conclusions

The NRC Safety Evaluation Report of September 11, 1975 (1) approved an ENC WREM-based PWR ECCS Evaluation Model consisting of generic aspects, approved once and referenced thereafter, and case particular facets to be presented for each plant. This approval was contingent upon the satisfactory completion of certain model solidification tasks. This Supplement addresses the items appropriate to the H. B. Robinson plant. The staff finds that the ENC WREM-based ECCS Evaluation Model is appropriate to the Robinson design model.

V. References

1. Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Pressurized Water Reactor ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model for Conformance to All Requrements of Appendix K to 10 CFR 50 by the Office of Nuclear Regulation, USNRC, September 11, 1975.

2. Letter: Report on Exxon Nuclear Company ECCS Evaluation Model for Reload Cores in PWR's, William Kerr, ACRS, to William Anders, NRC, September 15, 1975.

3. Evaluation of LOCA Hydrodynamics, Regulatory Staff: Technical Review, USAEC, November, 1974.

4. Unpublished Transmittal on Momentum Equations, K. V. Moore to NRC, September 11, 1975.

5. XN-75-41: Volume 1, ENC WREM-Based Generic PWR-ECCS-Evaluation Model, July 25, 1975.

6. XN-75-41: Supplement 6: Supplementary Information Relating to Blowdown and Heatup Analysis - Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model, October 27, 1975.

7. TN-188, Effects of Hot Walls on Flow in a Similated PWR Downcomer During a LOCA, Creare, Hanover, New Hampshire, May, 1974.

8. Piping Handbook, 5th Edition, Reno C. King McGraw-Hill, New York, 1967.

9. XN-75-41: Supplement 5, Revision 1, Supplementary Information Relating to Blowdown and Heatup Analysis, October 3, 1975.

10. XN-75-41: Supplement 7, Supplementary Information: ENC WREM-Based Generic PWR ECCS Evaluation Model, November 9, 1975.

11. Unpublished Transmittal on Calculation of Inlet Subcooling, ENC to NRC, November 20/21, 1975.

12. Water Reactor Evaluation Model (WREM): PWR Modalization and Sensitivity Studies, AEROJET Nuclear Company, July, 1974.

13. XN-75-41, Volume 2, Appendix D, ENC WREM-Based Generic PWR ECCS Evaluation Model, 3-Loop Westinghouse Large Break Example Problem (using September 26, 1975 Model), October 2, 1975.

Appendix B

1. "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors (PTS-PWR)", XN-74-5 Rev. 1, May 15, 1975.

2. "EXPOSE: The Exxon Nuclear Revised LEOPARD", XN-CC-21 Rev. 2, April 1975.

3. "XTG: A two-Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing (PWR Version)", XN-CC-28 Rev. 3, January 1975.

4. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR)", XN-74-5 Rev. 1, May 15, 1975.

5. "XTRAN-PWR: A Computer Code for the Calculation of Rapid Transients in Pressurized Temperature Feedback", XN-CC-32, October 7, 1975.

6. "Computational Procedure for Evaluating Fuel Rod Bowing (AXIBOW)," XN-75-32, April 1975.

7. "Definition and Justification of Exxon Nuclear Company DNB Correlation for PWR's," XN-75-48, October 6, 1975.

8. "XCOBRA-IIIC: A computer Code to Determine the Distribution of Coolant During Steady - State and Transient Core Operation," XN-75-21, April 1, 1975.

9. "Exxon Nuclear Company WREM - Based Generic PWR ECCS Evaluation Model," XN-75-41, Volumes I, II, III, Supplements 1-7.

10. Single Phase Hydraulic Performance of Westinghouse and Exxon Nuclear H. B. Robinson Fuel Assemblies XN-74-44, October 1974.

11. Technical Report on Densification of Exxon Nuclear PWR Fuels, U. S. Nuclear Regulatory Commission, February 27, 1975.

12. Densification Effects on Exxon Pressurized Water Reactor Fuel, XN-209, March 1974; (Including Supplements 1-4).

13. "Topical Report Power Distribution Control and Load Following Procedures", WCAP-8385, September, 1974.

14. "H. B. Robinson Unit 2 Justification for Operation at 2300 MWt." WCAP 8244, December, 1973.

15. "Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of CP & L H. B. Robinson Steam Electric Plant Unit No. 2 Power Increase," Basic Report dated May 20, 1974, Supplement No. 1 dated July 31, 1975.

16. "Plant Transient Analysis for H. B. Robinson Unit 2 PWR for 2300 MWt", XN-75-14, July 15, 1975.

17. "Control Rod Ejection Accident for H. B. Robinson Unit 2 Based on Exxon Nuclear Reload Fuel", XN-75-44, July 25, 1975.

18 Letter from R. W. Reid, NRC to J. A. Jones, CP & L dated October 17, 1975.

19. "Safety Evaluation Report Regarding Review of the Exxon Nuclear Company Pressurized Water Reactor ECCS Codes and the H. B. Robinson Reactor ECCS Evaluation Model for Conformance to all Requirements of Appendix K to 10 CFR 50 by the Office of Nuclear Reactor Regulation", USNRC, September 11, 1975.

20. "H. B. Robinson Unit No. 2 LOCA Analyses Using the ENC WREM Based PWR ECCS Evaluation Model (September 26, 1975 version)", XN-75-57, Revision 1, November 9, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Facility Operating License No. DPR-23 issued to Carolina Power & Light Company which revised Technical Specifications for operation of the H. B. Robinson Unit 2, located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment establishes operating limits in the Technical Specifications based upon an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR § 50.46, and revises provisions related to the replacement of 52 fuel assemblies in the Robinson-2 core with fuel assemblies of a different design, constituting refueling of the core for operation with Cycle 4.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on October 31, 1975 (40 F.R. 50753). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated October 14, 1975, as supported by correspondence dated August 3, August 22, October 17, November 13, November 18, and November 24, 1975, (2) Amendment No. 15 to License No. DPR-23, with Change No. 40, (3) the Commission's related Safety Evaluation, (4) the Commission's Negative Declaration dated December 2, 1975 which is being published concurrently with this notice, and (5) the Commission's associated Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Hartsville Memorial Library, Home & Fifth Avenues, Hartsville, South Carolina.

A copy of items (2), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 3rd day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Btanch #4
Division of Reactor Licensing

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-23

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

DOCKET NO. 50-261

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License No. DPR-23. These changes would authorize the Carolina Power and Light Company (the licensee) to operate the H. B. Robinson Steam Electric Plant, Unit 2 (located in Darlington County, Hartsville, South Carolina), with limiting conditions associated with fuel assembly specific power resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS).

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License No. DPR-23, H. B. Robinson Steam Electric Plant, Unit 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statement for H. B. Robinson Steam Electric Plant, Unit 2, published in April 1975.

The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina.

Dated at Rockville, Maryland, this 2nd day of December 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Wm. H. Regan

William H. Regan, Jr., Chief
Environmental Projects Branch #4
Division of Reactor Licensing

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Facility Operating License No. DPR-23 issued to Carolina Power & Light Company which revised Technical Specifications for operation of the H. B. Robinson Unit 2, located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment establishes operating limits in the Technical Specifications based upon an evaluation of ECCS performance calculated in accordance with an acceptable evaluation model that conforms to the requirements of the Commission's regulations in 10 CFR § 50.46, and revises provisions related to the replacement of 52 fuel assemblies in the Robinson-2 core with fuel assemblies of a different design, constituting refueling of the core for operation with Cycle 4.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on October 31, 1975 (40 F.R. 50753). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated October 14, 1975, as supported by correspondence dated August 3, August 22, October 17, November 13, November 18, and November 24, 1975, (2) Amendment No. 15 to License No. DPR-23, with Change No. 40, (3) the Commission's related Safety Evaluation, (4) the Commission's Negative Declaration dated December 2, 1975, which is being published concurrently with this notice, and (5) the Commission's associated Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Hartsville Memorial Library, Home & Fifth Avenues, Hartsville, South Carolina.

A copy of items (2), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 3rd day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Btanch #4
Division of Reactor Licensing