Docket No. 50-261

JUL 25 1973

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R. Woodruff (2)
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S. Kari
V. Stello
J. Hendrie
AEC PDR
Local PDR

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

Gentlemen:

Because of the lack of adequate reserve power generating capability in the area served by Carolina Power & Light Company and their neighboring utilities, you requested by letters dated June 20 and July 6, 1973, authority to operate Unit No. 2 at the H. B. Robinson Steam Electric Plant at levels up to 100% of full power in lieu of the present maximum authorized level of 94.8% of full power.

As discussed with members of your staff, certain changes to your proposed conditions for operation at levels up to 100% of full power are necessary to meet our regulatory requirements. These revised conditions are enclosed. We conclude that adherence to these conditions will provide reasonable assurance in the event of a loss-of-coolant accident that the temperature of fuel clad in Regions 2 and 3 would not exceed 1800°F and in Region 4 would not exceed 2300°F.

Operation of the Unit No. 2 in the manner described and at power levels up to and including 100% of licensed power will allow an adequate margin of safety. Operation in this manner is authorized thru the predicted peak summer demand for power.

Sincerely,

Original signed by:
Roger S. Boyd

for A. Giambusso
Deputy Director for
Reactor Projects
Directorate of Licensing

Enclosure:
Interim Conditions for Operation

OFFICE ▶	L:ORB#1	L:ORB#1	L:RS	L:TR	L:OR	L:RP
SURNAME ▶	RWWoodruff:sep	RJSchemel	VStello	JMHendrie	DJSkovholt	AGiambusso
DATE ▶	7/20/73	7/23/73	7/ /73	7/[illegible]/73	7/[illegible]/73	7/24/73

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43-16-81465-1 445-678

INTERIM CONDITIONS FOR OPERATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

1. The operational mode of the plant will be base load, with the exception of the weekly turbine valve test.

2. A full-core flux map utilizing the movable detector system will be performed at least every two weeks to determine the value of the maximum peak kW/ft. The value so determined shall be maintained $\leq$ 14.2 kW/ft in Regions 2 and 3 and $\leq$15.1 in Region 4.

3. Surveillance of the axial peaking factor will be performed every eight hours with at least two thimbles, representative of the average core axial peaking factor, F_z. The limiting value of F_z will be determined in the following manner:

$$F_z = \frac{F_q^T}{F_{xy}^N \cdot F_q^E \cdot F_u^N \cdot S(z) \cdot P}$$

where F_q^T is the maximum core peaking factor, F_{xy}^N is the radial peaking factor in the plane of the hot spot, F_q^E is the engineering uncertainty factor, F_u^N is the nuclear uncertainty factor, S(z) is the spike penalty factor, and P is the fraction of licensed power. The values to be used for F_q^E and F_u^N are 1.03 and 1.09 respectively. The value for F_u^N is the product of 1.04 and 1.05 where 1.04 is the uncertainty allowance for axial surveillance and 1.05 is the uncertainty allowance for determining F_{xy}^N. For Regions 2 and 3, the values to be used for the remaining peaking factors are 2.41 for F_q^T and 1.40 for F_{xy}^N. For Region 4, the values to be used are 2.56 for F_q^T and 1.51 for F_{xy}^N. The values selected for S(z) will be obtained from the attached Figure. The selected values will be at or above the value of z for which the product of the axial traverse and S(z) is maximum.

4. The movement of Bank D during power operations will be restricted to no more than a total of five steps in any one direction with the above surveillance frequency. If the rod bank movement exceeds five steps, the frequency of surveillance will be increased to monitor F_z at the design intervals for the Axial Power Distribution Monitoring System or the equivalent.

5. Surveillance of F_z will be performed during and after testing of the turbine trip valves.

6. The APDMS and the plant computer shall be employed in the surveillance of F_z. If the APDMS or plant computer must be taken out of service, F_z surveillance may be performed manually provided that an additional person appropriately trained is added to each crew until the APDMS and the plant computer are returned to service. If manual surveillance is not performed, power must not exceed 94.8% of licensed power.

7. The limiting value of F_z during surveillance will be the lower of the two values as determined from Section 3. Thimbles at core positions L4, D5, D12, or F-13 or equivalent shall be used for surveillance.

8. Reactor shutdown will be initiated within 8 hours if the primary to secondary leakage in a steam generator reaches 0.3 gpm.

9. a. Primary coolant gross radioactivity shall be measured at least five times per week and after each significant operating event which could affect fuel clad integrity.

   b. Primary coolant gross gamma radioactivity shall be monitored continuously by the letdown monitor. If the letdown monitor is not operating, the primary coolant gross activity shall be measured daily.

   c. Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured at least daily to evaluate steam generator leak tightness.

10. A biweekly report of the results of incore surveillance and a monthly report of all primary and secondary activity measurements and effluent discharge activity levels shall be made to the Directorate of Licensing.

11. The following administrative controls shall be applied:

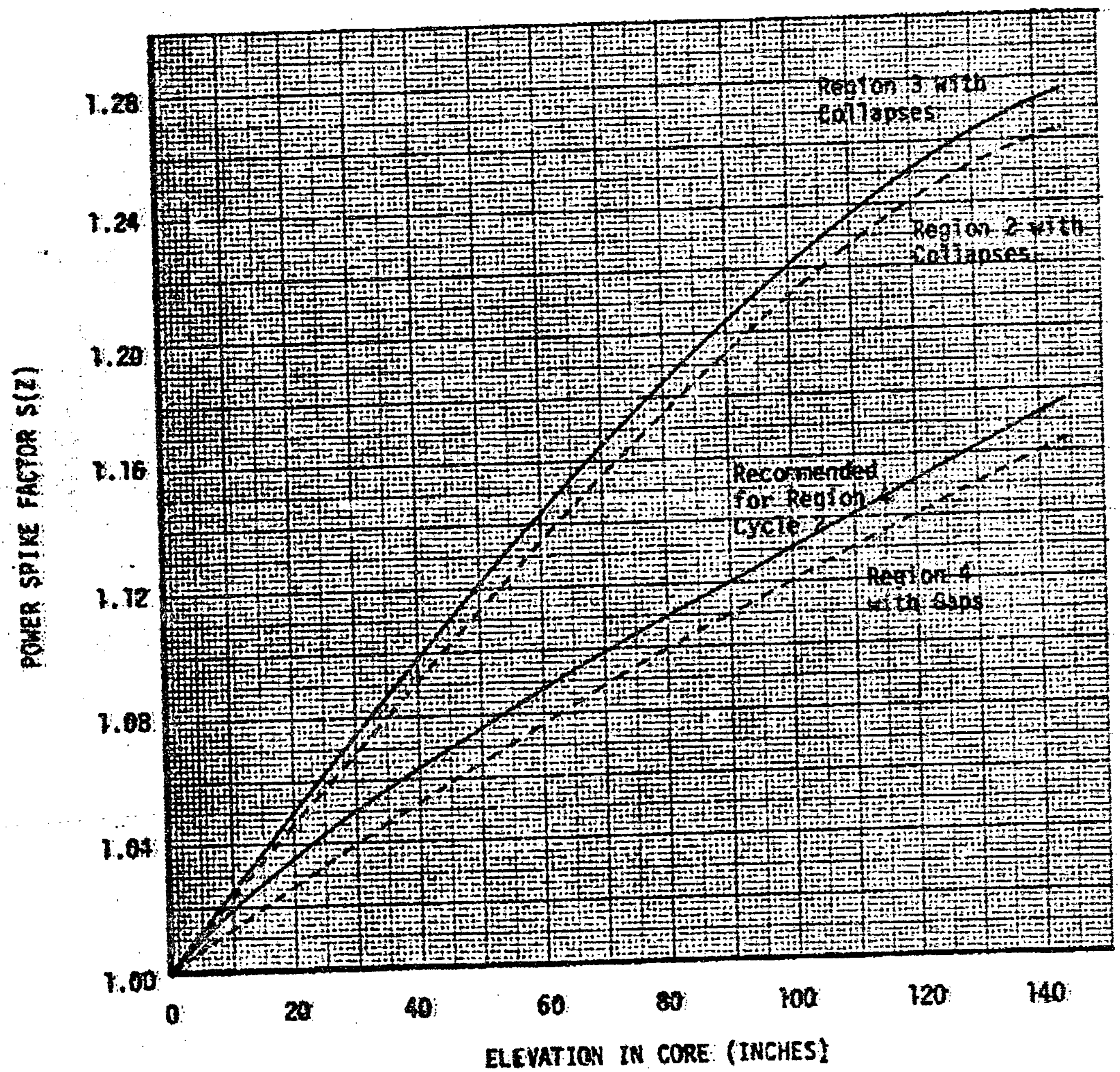
    a. Should a power level less than 95% be maintained continuously for more than 100 hours but less than 24 days, the rate of power increase shall be limited to 10% per hour.

    b. Should a power level less than 95% be maintained continuously for more than 24 days, the rate of power increase shall be limited to 3% per hour from 25% to 100% of full power.

FIGURE

POWER SPIKE VS ELEVATION

H. B. ROBINSON UNIT 2 - CYCLE 2



JUL 11 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

A copy of our "Notice of Opportunity for Hearing Pursuant to 10 CFR Part 50, Appendix D, Section B" dated July 6, 1973, that has been forwarded to the Office of the Federal Register for publication on July 18, 1973, is enclosed for your information.

As required by AEC regulations, the notice provides 30 days for filing a request for a hearing or for filing a petition for leave to intervene in the environmental review process for the H. B. Robinson Unit 2.

A copy of a related display ad that is being transmitted to the Florence Morning News and Columbia State papers for publication on July 18, 1973, also enclosed for your information.

Sincerely,

Original signed by
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosures:
1. Federal Register Notice
2. Newspaper Ad

cc w/enclosures:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C.

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

See additional cc's on next page

JUL 11 1973

cc w/enclosures:
Mr. J. Bonner Manly, Director
State Development Board
Hampton Office Building
Columbia, South Carolina 29202

Chairman
Darlington County Commission
County Court House
Darlington, South Carolina 29532

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N. Dube
D. Skovholt
T. Carter
R. Schemel
R. Woodruff(2)
S. Teets

OFFICE ▶	L:ORB#2 x-7403	L:ORB#1	OGC concurred in notice w/	L:ORB#1		
SURNAME ▶	RMDiggs:dc	RWWoodruff	memo to Secy.	RJSchemel		
DATE ▶	7/10/73	7/11/73	dtd 7/9/73	7/11/73		

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 11, 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

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A copy of a related display ad that is being transmitted to the Florence Morning News and Columbia State papers for publication on July 18, 1973, also enclosed for your information.

Sincerely,

Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosures:
1. Federal Register Notice
2. Newspaper Ad

cc w/enclosures:
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C.

Hartsville Memorial Library
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Hartsville, South Carolina 29550

See additional cc's on next page

cc w/enclosures:
Mr. J. Bonner Manly, Director
State Development Board
Hampton Office Building
Columbia, South Carolina 29202

Chairman
Darlington County Commission
County Court House
Darlington, South Carolina 29532

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-261
	)	
(H. B. Robinson Unit No. 2)	)	

NOTICE OF OPPORTUNITY FOR HEARING PURSUANT TO 10 CFR PART 50 APPENDIX D, SECTION B

The Carolina Power and Light Company (the licensee) is the holder of Operating License No. DPR-23 (the operating license), issued by the Atomic Energy Commission on July 31, 1970. The operating license authorizes the licensee to possess, use, and operate a pressurized water nuclear reactor, designated as the H. B. Robinson Unit No. 2, at steady-state power levels up to a maximum of 2200 megawatts (thermal) at the licensee's site in Darlington County, South Carolina, in accordance with technical specifications appended thereto.

The facility is subject to the provisions of Section B of Appendix D to 10 CFR Part 50, which sets forth procedures applicable to review of environmental considerations for production and utilization facilities for which construction permits or operating licenses were issued in the

period January 1, 1970 - September 9, 1971. Notice is hereby given, pursuant to the Commission's "Rules of Practice," in 10 CFR Part 2, and Appendix D of 10 CFR Part 50, "Implementation of the National Environmental Policy Act of 1969," that the Commission is providing an opportunity for hearing with respect to whether, considering those matters covered by Appendix D to 10 CFR Part 50, the existing full-term facility operating license should be continued, modified, terminated or appropriately conditioned to protect environmental values. Requests for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's rules of practice in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed within the time prescribed in this notice, the Commission or an Atomic Safety and Licensing Board designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel will rule on the request and/or petition and the Secretary of the Commission or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

A petition for leave to intervene must be filed under oath or affirmation in accordance with the provisions of 10 CFR §2.714. As required by 10 CFR §2.714, a petition for leave to intervene shall set forth the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and any other contentions

of the petitioner including the facts and reasons why he should be permitted to intervene, with particular reference to the following factors: (1) the nature of the petitioner's right under the Atomic Energy Act of 1954, as amended ("the Act"), to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. Any such petition shall be accompanied by a supporting affidavit identifying the specific aspect or aspects of the subject matter of the proceeding as to which the petitioner wishes to intervene and setting forth with particularity both the facts pertaining to his interest and the basis for his contentions with regard to each aspect on which he desires to intervene. A petition that sets forth contentions relating only to matters outside the jurisdiction of the Commission will be denied.

A request for a hearing or a petition for leave to intervene must be filed either by mail with the Office of the Secretary of the Commission, United States Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Staff, or by delivery to the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., not later than thirty (30) days from the date of publication of this notice in the FEDERAL REGISTER. A petition for leave to intervene which is not timely will not be granted unless the Commission, the presiding officer, or the Atomic Safety and Licensing Board designated to rule on the petition or request determines that the petitioner

has made a substantial showing of good cause for failure to file on time and after the consideration of those factors specified in 10 CFR Part 2, Section 2.714(a).

For further details with respect to the matters under consideration, see (1) the licensee's Environmental Report dated November 5, 1971, and (2) the Commission's draft environmental statement on environmental considerations pursuant to 10 CFR Part 50, Appendix D, dated April 1973, both of which 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 and Fifth Avenues, Hartsville, South Carolina 29550.

The Commission's final detailed statement on environmental considerations will also be available at the above locations upon issuance. A copy of this item may be obtained when available by request to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Director of Licensing.

Dated at Bethesda, Maryland, this 6th day of July 1973.

FOR THE ATOMIC ENERGY COMMISSION

L. Burger

A. Burger, Acting Chief
Operating Reactors Branch #1
Directorate of Licensing

NOTICE OF OPPORTUNITY FOR PUBLIC PARTICIPATION

IN AEC LICENSING PROCEEDING FOR

H. B. ROBINSON UNIT 2

The Atomic Energy Commission is giving public notice that it is providing opportunity for a hearing on whether the full-term facility operating license held by Carolina Power & Light Company for operation of the H. B. Robinson Unit 2, located in Darlington County, South Carolina, should be continued, modified, terminated, or appropriately conditioned to protect environmental values.

The notice provides that within 30 days after publication in the Federal Register on July 18, 1973, any person whose interest may be affected may file a petition for leave to intervene in the proceeding.

Petitions for leave to intervene must be filed under oath or affirmation and must set forth the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, and the petitioner's contentions with respect to the proposed licensing action. Such petitions must be filed in accordance with the above referenced Federal Register notice and must be filed with the Secretary of the Commission, U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Chief, Public Proceedings Staff by 1973. A copy of the petition and/or request for hearing should be sent to the Chief Hearing Counsel, Office of the General Counsel, U. S. Atomic Energy Commission, Washington, D. C. 20545 and to George F. Trowbridge, Esquire, Shaw, Pittman, Potts, Trowbridge and Madden, 910 - 17th Street, N. W., Washington, D. C. 20006, attorney for the licensee.

A petition for leave to intervene must be accompanied by a supporting affidavit that identifies the specific aspect or aspects of the proceedings as to which the petitioner relies as to both his interest and his contentions with regard to each aspect on which intervention is requested. Petitions stating contentions relating only to matters outside the Commission's jurisdiction will be denied.

All petitions will be acted upon by the Commission or designated licensing board or by the Chairman of the Atomic Safety and Licensing Board Panel. Timely petitions will be considered to determine whether a hearing should be noticed or another appropriate order issued regarding the disposition of the petitions.

In the event that a hearing is held and a person is permitted to intervene, he becomes a party to the proceeding and has a right to participate fully in the conduct of the hearing. For example, he may present evidence and examine and cross-examine witnesses.

A copy of the Federal Register notice is on file for public inspection at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29550, and the Commission has arranged for other documents and correspondence relating to the licensing of this facility to be kept at the same location.

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RJSchemel
DJSkovholt
TJCarter
SATeets
RWWoodruff

July 9, 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

Your letter of June 13, 1973, informed us that the additional time and expense is not justified in furnishing the additional information we requested in order to complete our review of your proposed change to the Technical Specifications to revise control rod insertion limits. Accordingly, we have discontinued our review of the proposed change and consider it withdrawn per your June 13 letter.

Sincerely,

Original Signed by.
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 -17th Street, N. W.
Washington, D. C. 20006

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Home & Fifth Avenues
Hartville, South Carolina 29550

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SURNAME ▶	SATeets:lb	RWWoodruff	RJSchemel	DJSkovholt		
DATE ▶	7/3/73	7/3/73	7/3/73	7/9/73		

JUL 05 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley
Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

In your letter of May 25, 1973, you reported a performance check on the Safety Injection pumps in which several data points fell below the minimum performance curve. The minimum performance curve used in this case was one established August 12, 1970, which modified the requirements in the FSAR.

Figure 1 of your May 25, 1973, report also indicates deteriorating pump performance. The margin between current and minimum performance has essentially disappeared.

We consider your decision to have Westinghouse Electric Corporation and Worthington Pump International Corporation review the result of these system and pump tests and your offer to forward additional comments and recommendations to be appropriate.

In making these recommendations and comments, please include the following:

1. A comprehensive report describing the present status of the Safety Injection System (SIS) and the chronology that led to this condition, including references to the FSAR and previous correspondence.

2. Summarize your review of the accident analyses to determine which accidents are most significant with respect to the cooling and boron injection functions of the SIS. For the loss-of-coolant accident, use 1800°F for the limit on fuel cladding temperature.

   Based on the above information, determine a minimum performance curve for the SIS pumps.

3. Provide a summary of test procedures used to assure both the SIS and pumps meet minimum requirements. Include a schema showing valve and instrumentation set up used in performing these tests.

Sincerely,

A. Burger for

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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OGC
R. J. Schemel
L. McDonough
S. Teets
T. J. Carter

OFFICE ▶	L:ORB#1	L:ORB#1	L:ORB#1			
SURNAME ▶	x-7434 LMcDonough:dc	STeets	A. Burger RJSchemel			
DATE ▶	7/3/73	7/3/73	7/5/73			

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RWWoodruff
SATeets
DRoss

JUN 25 1973

Docket No. 50-261

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

Gentlemen:

By letter dated June 20, 1973, you requested authority to operate the Unit No. 2 at the H. B. Robinson Steam Electric Plant at 100% of licensed power pending issuance of changes to the Technical Specifications based on fuel densification. Having reviewed your letter and other pertinent information, we conclude that the additional information identified in Attachment A is required before we can act on your request.

Sincerely,

Original signed by
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosure:
Attachment A - Request for
Additional Information

cc w/enclosure:
G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden 910 - 17th Street, N. W.
Washington, D. C. 20006

OFFICE ▶	L:ORB#1	L:ORB#1	L:CPB	L:ORB#1		
	x-7434		VStello DR for VS			
SURNAME ▶	RWWoodruff:dc	SATeets	DRoss	RJSchemel		
DATE ▶	6/25/73	6/25/73	6/25/73	6/25/73		

ATTACHMENT A

REQUEST FOR ADDITIONAL INFORMATION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

1. Reanalyze the limiting loss-of-coolant accident assuming that the axial power distribution and the corresponding spike penalty factor are such that maximum fuel cladding temperature is obtained. Submit the results of the analysis and a discussion of the significance of the results with respect to incore monitoring of the power distribution.

2. For the limiting axial power distribution described above, reanalyze the most significant anticipated transient to determine the minimum departure from nucleate boiling ratio which results. Submit the results, and if the results indicate, propose a limit for the enthalpy rise hot channel factor.

3. Propose requirements for monitoring and limiting the shape of the axial power distribution. Identify the criteria for selecting the two thimbles that are to be used for surveillance of the axial power distribution.

4. With regard to both automatic and manual surveillance of the axial power distribution, provide and justify appropriate values for the nuclear uncertainty hot channel factor and for the frequency of surveillance.

5. Justify using 1.15 for spike penalty factor in Region 4 assuming that the peak pin in this region is adjacent to Region 3 or use 1.25 for the spike penalty factor.

6. Perform incore surveillance of the axial power distribution when the reactor is operating above 75% of licensed full power or provide a correlation of the heat flux hot channel factor to axial offset which is based on appropriate values of the radial peaking factor. If the latter is elected, describe the operating manuevers assumed in preparing the correlation.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket File

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

June 15, 1973

Docket No. 50-261

TO ALL RECIPIENTS

Please substitute the attached page entitled "Interim Conditions for Operation - H. B. Robinson Unit No. 2" for the first page of the attachment to our letter of June 5, 1973, to Carolina Power & Light Company, License No. DPR-23.

INTERIM CONDITIONS FOR OPERATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

1. The peak linear power density, as obtained from evaluations of in-core flux maps, shall not exceed 14.5 kW/ft.

2. Reactor shutdown will be initiated within 8 hrs if the primary to secondary leakage in a steam generator reaches 0.3 gpm.

3. For Cycle 1 operation:

   (a) In-core flux traces shall be measured and evaluated every 2 weeks. A full in-core flux map shall be measured and evaluated once a month.

   (b) Primary coolant gross radioactivity shall be measured at least five times per week and after each significant operating event which could affect fuel clad integrity.

   (c) Primary coolant gross gamma radioactivity shall be monitored continuously by the letdown monitor. If the letdown monitor is not operating, the primary coolant gross activity shall be measured daily.

   (c) Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured at least daily to evaluate steam generator leak tightness.

4. A monthly report of all primary and secondary activity measurements and effluent discharge activity levels shall be made to the Directorate of Licensing.

5. For Cycle 1 operation, the following administrative controls shall be applied:

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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JUN 14 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

Pending completion of our evaluation of the effect of fuel densification on the performance of Unit No. 2 at the H. B. Robinson Steam Electric Plant and issuance of changes to the Technical Specifications, as appropriate, we requested that you operate Unit No. 2 at levels less than 94.8% of licensed full power. However, because of present regional power needs, you requested authority by letter dated June 13, 1973, to operate Unit No. 2 through the peak load on Friday, June 15, 1973, at levels up to 100% of licensed full power.

During this period, you would perform additional surveillance which would assure that the peak linear power density including the effect of local power spikes would be less than 14.2 kW/ft. For this power density, information which you have derived using methods that we have found acceptable indicates that the temperature of flattened fuel clad would not exceed 1800°F in the event of a loss of coolant accident.

We conclude that operation of Unit No. 2 in the manner proposed will provide an acceptable margin of safety.

Sincerely,

Original signed by:
Roger S. Boyd

for A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

cc: see next page

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43-16-81465-1 445-678

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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OFFICE ▶	L:ORB#1	L:ORB#1	L:RS	L:TR	L:OR	L:RP
SURNAME ▶	SATeets RWWoodruff:dd	RJSchemel	VStello	JMHendrie	DJSkovholt	AGiambusso
DATE ▶	6/13/73	6/13/73	6/13/73	6/13/73	6/14/73	6/14/73

JUN 5 1973

Docket No. 50-261

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

Gentlemen:

In our letter dated May 7, 1973, you were advised that Unit No. 2 at the H. B. Robinson Steam Electric Plant should not be operated at levels above 75% of licensed power pending completion of our review of the effect of fuel densification on Cycle 2 operations. Because of the increasing demand for power expected in your area due to the onset of summer, you requested by letter dated June 4, 1973, that this temporary limit on power level be changed to 94% of licensed power. Your request is based on the similarity of Robinson-2 and Point Beach-1 systems and fuel cycles and on our action dated May 22, 1973, permitting resumption of operation of Point Beach-1 at full power.

We are currently reviewing the analysis of the effect of fuel densification on Cycle 2 for Robinson-2 and proposed changes to the Technical Specifications which were submitted in your letter dated April 20, 1973. Although our review is not completed, it has progressed to the point where we conclude that the analyses of fuel performance, power capability, and postulated accidents are substantially similar for Robinson-2 and Point Beach-1. These analyses indicate that our interim criteria for fuel clad temperature will be satisfied provided that Robinson-2 is operated at less than 94.8% of licensed power with peak linear power density at less than 14.5 kW/ft.

We conclude that operation of Robinson-2 in accordance with the revised Interim Conditions for Operation (attached) and in accordance with

the conditions proposed in your letter of June 4, 1973, will provide an acceptable margin of safety for Cycle 2 operation pending completion of our review.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
Interim Conditions for Operation

cc w/enclosure:
George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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V. Stello
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OFFICE ▶	L:ORB#1 RWW	L:ORB#1	L:OR	L:RS	L:TR	L:RP
x-7434 SURNAME ▶	RWWoodruff:dc SATeets	RJSchemel	DJSkovholt	VStello	JMHendrie	AGiambusso
DATE ▶	6/5/73	6/5/73	6-5/73	6/5/73	6/6/73	6/6/73

INTERIM CONDITIONS FOR OPERATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

1. The peak linear power density, as obtained from evaluations of in-core flux maps, shall not exceed 14.5 kW/ft.

2. Reactor shutdown will be initiated within 8 hrs if the primary to secondary leakage in a steam generator reaches 0.3 gpm.

3. For Cycle 2 operation:

   (a) In-core flux traces shall be measured and evaluated every 2 weeks. A full in-core flux map shall be measured and evaluated once a month.

   (b) Primary coolant gross radioactivity shall be measured at least five times per week and after each significant operating event which could affect fuel clad integrity.

   (c) Primary coolant gross gamma radioactivity shall be monitored continuously by the letdown monitor. If the letdown monitor is not operating, the primary coolant gross activity shall be measured daily.

   (d) Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured at least daily to evaluate steam generator leak tightness.

4. A monthly report of all primary and secondary activity measurements and effluent discharge activity levels shall be made to the Directorate of Licensing.

5. For Cycle 2 operation, the following administrative controls shall be applied:

(a) Should a power level less than 95% be maintained continuously for more than 100 hours but less than 24 days, the rate of power increase shall be limited to 10% per hour.

(b) Should a power level less than 95% be maintained continuously for more than 24 days, the rate of power increase shall be limited to 3% per hour from 25% to 100% of full power.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

JUN 0 5 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

By letters dated April 24 and May 1, 1973, you reported two abnormal occurrences involving the uncontrolled release of radioactive liquid from Unit No. 2 at the H. B. Robinson Steam Electric Plant. So that we can complete our evaluation of your reports, it will be necessary for us to have the additional information identified in Attachment A. Please submit this information by June 30, 1973.

Sincerely,

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosure:
Attachment A - Request for Information

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RJSchemel
TJCarter

cc: G. F. Trowbridge, Esquree
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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OFFICE ▶	L:ORB #1	L:ORB #1	L:ORB #1			
SURNAME ▶	RWWoodruff:lb	SATeets	RJSchemel			
DATE ▶	5/23/73	5/24/73	6/5/73			

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43-16-81465-1 445-678

ATTACHMENT A

REQUEST FOR INFORMATION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

1. For the April 10 abnormal occurrence, provide a schematic diagram of the system in which the leak occurred and of related systems. Identify on the schematic diagram: design pressures, the valve which leaked, the safety injection pump which was running, the spectacle flange and any other flanges of interest. Describe the need for: interconnection of the systems involved, the spectacle flange and any similar or related flanges, and pressure relief valves and appropriate routing of their effluents.

2. For spaces containing the systems above, describe in detail: their drainage, the location and volume of sumps, sump level sensors and alarms, and the divide between areas which drain to onsite and offsite sumps.

3. For the abnormal occurrences of April 10 and 23, identify in detail the path of the effluent from the source of leakage to Black Creek. Indicate the points at which the effluent crossed restricted area boundaries, the fence line, and the site boundary. Identify the points at which dilution occurred and how the amount of dilution was estimated. Estimate the fraction of MPC (10 CFR Part 20, Appendix B, Table II, Column 2) at each point. Provide the results of soil samples taken from the effluent path.

4. Identify all other tanks and systems which would release radioactive liquid from the site in the event of overflow or leakage.

5. For the tanks and systems above, describe possible modifications which would direct all radioactive liquid overflows to the radioactive liquid waste system and all radioactive liquid spills to sumps of appropriate capacity onsite.

MAY 07 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Vice President
Bulk Power Supply
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

In response to our letter of November 20, 1972, the Carolina Power & Light Company submitted its analysis of the effects of fuel densification on Cycle 2 operation of Unit No. 2 at the H. B. Robinson Steam Electric Plant (Westinghouse Report WCAP-8115, non-proprietary) on April 20, 1973. The analysis follows the methods previously presented for Point Beach Unit No. 2 and reviewed by the staff, and in addition takes into account expected fuel clad flattening in regions 2 and 3 of the Cycle 2 core. You have concluded from the analysis that full power operation is justified with appropriate provisions made for reduced power peaking and for limiting steam generator leakage.

While the Regulatory staff review of the H. B. Robinson Unit No. 2 Cycle 2 operations has been initiated, it is not expected to be completed for several weeks because of review commitments regarding fuel densification effects at other facilities. We advised you of this during a discussion at our offices on April 24, 1973. Pending completion of the Regulatory staff review, you should limit operation of Unit No. 2 to a power level no greater than 75% of licensed full power. Further, you should continue to observe the special operating instructions imposed during the latter part of Cycle 1 (enclosed as an attachment to our letter of September 12, 1972).

We have concluded that operation of H. B. Robinson Unit No. 2 in accordance with these restrictions will provide an acceptable margin of safety for Unit No. 2 Cycle 2 operation pending completion of our review. Our intention is to complete this review as soon as possible consistent with

our other commitments. In the interim, operation of the plant should be in accord with limitations listed above unless our approval of modifications to the limitations has been obtained.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
Safety Evaluation

cc w/enclosure:
George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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T. Carter
R. Woodruff (2)
S. Teets
V. Stello
J. Hendrie

SEE PREVIOUS YELLOW FOR OTHER CONCURRENCES

OFFICE ▶	L:ORB#1	L:ORB#1	L:OR	L:RP		
SURNAME ▶	RWWoodruff:dc SATeets	RJSchemel	DJSkovholt	AGiambusso		
DATE ▶	5/7/73	5/ /73	5/ /73	5/7/73		

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SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

By letter dated April 20, 1973, Carolina Power & Light Company proposed operation of Robinson-2 Cycle 2 at power levels up to 100% of licensed power but with limitations on the power distribution in order to limit the fuel cladding temperature in the event that a loss-of-coolant accident were to occur. The proposed limitations are based on the potential for collapse of the fuel cladding.

The Regulatory staff has recently completed its review of fuel densification for Surry 1 and 2 Cycle 1. The staff concludes that these three-loop units can be operated at 15.2 kW/ft and that in the event of a LOCA the fuel cladding temperature would not exceed 2300°F. Because Robinson-2 is a three-loop unit, essentially the same relationship between linear power density and fuel cladding temperature will pertain.

Cladding collapse is not expected in Surry 1 and 2 Cycle 1 but is expected in Robinson-2 Cycle 2. For this reason, we conclude that the post-LOCA cladding temperature for Robinson-2 Cycle 2 should be limited to 1800°F. The Regulatory staff considers that a 25% reduction in linear power density is adequate in this regard. Therefore, pending completion of the Regulatory staff review of the Robinson-2 Cycle 2 densification analysis, we conclude that linear power density, as determined by flux mapping, must be limited to 11.4 kW/ft. Further, the values of flux taken from the maps must be increased by a factor of 1.2 because of the possibility of undetected flux blips.

We conclude that operation of Robinson-2 Cycle 2 in accordance with these restrictions will provide an acceptable margin of safety pending completion of the Regulatory staff's review.

Nevertheless, pending completion of the Regulatory staff review of the densification analysis by Carolina Power & Light Company for the Robinson-2, we are temporarily limiting operation of Robinson-2 to 75% of licensed power.

R. W. Woodruff
Operating Reactors Branch #1
Directorate of Licensing

OFFICE ▶			Robert J. Schemel, Chief			
			Operating Reactors Branch #1			
SURNAME ▶			Directorate of Licensing			
DATE ▶ Date:	MAY 07 1973					

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Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Vice President
Bulk Power Supply
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

In response to our letter of November 20, 1972, the Carolina Power & Light Company submitted its analysis of the effects of fuel densification on Cycle 2 operation of Unit No. 2 at the H. B. Robinson Steam Electric Plant (Westinghouse Report WCAP-8115, non-proprietary) on April 20, 1973. The analysis follows the methods previously presented for Point Beach Unit No. 2 and reviewed by the staff, and in addition takes into account expected fuel clad flattening in regions 2 and 3 of the Cycle 2 core. You have concluded from the analysis that full power operation is justified with appropriate provisions made for reduced power peaking and for limiting steam generator leakage.

While the Regulatory staff review of the H. B. Robinson Unit No. 2 Cycle 2 operations has been initiated, it is not expected to be completed for several weeks because of review commitments regarding fuel densification effects at other facilities. We advised you of this during a discussion at our offices on April 24, 1973. Pending completion of the Regulatory staff review, we request that you limit operation of Unit No. 2 to a power level no greater than 75% of licensed full power. Further, we request that you continue to observe the special operating instructions imposed during the latter part of Cycle 1 (enclosed as an attachment to our letter of September 12, 1972).

We have concluded that operation of H. B. Robinson Unit No. 2 in accordance with these restrictions will provide an acceptable margin of safety for Unit No. 2 Cycle 2 operation pending completion of our review. Our intention is to complete this review as soon as possible consistent with our other commitments.

Sincerely,

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

OFFICE ▶
cc: see next page

FILE THIS MRS WITH YELLOW COPY

MEMO ROUTE SLIP
Form AEC-93 (Rev. May 14, 1947) AECM 0240

- [ ] See me about this.
- [ ] Note and return.
- [ ] For concurrence.
- [ ] For signature.
- [ ] For action.
- [ ] For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
Angie		Woodruff's reasons for wanting 89 % are good ones; however:
TO (Name and unit)	INITIALS / DATE	REMARKS — our overall regulatory posture on fuels problems can not stand anything but the
TO (Name and unit)	INITIALS / DATE	REMARKS usual 75 %, pending completion of our review of operation with collapsed fuel

FROM (Name and unit): Hendrie

PHONE NO.

DATE 5/4

REMARKS rods (for which Pt Beach 1 is model). (Also, the tech spec on peaking factor surveillance for 89 % would be difficult to work out, & worse to operate with.)

USE OTHER SIDE FOR ADDITIONAL REMARKS

GPO : 1971 O - 445-469

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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R. Schemel
R. Carter
R. Woodruff (2)
S. Teets
V. Stello

I do not concur for the following reasons:
1. Linear power density, not power, is the significant parameter;
2. 11.4 kW/ft is an adequate limit and would permit operation at up to 89% of full power;
3. At peak summer load which might occur in June, the predicted reserve for the Virginia Carolinas Area is a marginal 10.5%. This does not include limits on Surry 1 & 2 and Robinson-2 which would reduce the reserve to 8.8%.
An alternate action letter is attached.
R. W. Woodruff
5/3/73

I have considered the above comments. I agree with no. 1. No 2 has technical merit but I believe that the uncertainties currently existing with fuel densification justify the added conservatism of a 75% power limit.
5/4

OFFICE ▶	L:ORB#1	L:ORB#1	L:RS	OGC	L:OR	L:RP
SURNAME ▶	RWWoodruff:dc ~~SATeets~~	RJSchemel	VStello	5/4/73	DJSkovholt	AGiambusso
DATE ▶	5/3/73	5/7/73	5/4/73	5/ /73	5/4/73	5/7/73

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Vice President
Bulk Power Supply
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

In response to our letter of November 20, 1972, the Carolina Power & Light Company submitted its analysis of the effects of fuel densification on Cycle 2 operation of Unit No. 2 at the H. B. Robinson Steam Electric Plant (Westinghouse Report WCAP-8115, non-proprietary) on April 20, 1973. The analysis follows the methods previously presented for Point Beach Unit No. 2 and reviewed by the staff, and in addition takes into account expected fuel clad flattening in regions 2 and 3 of the Cycle 2 core. You have concluded from the analysis that full power operation is justified with appropriate provisions made for reduced power peaking and for limiting steam generator leakage.

While the Regulatory staff review of the H. B. Robinson Unit No. 2 Cycle 2 operations has been initiated, it is not expected to be completed for several weeks because of review commitments regarding fuel densification effects at other facilities. We advised you of this during a discussion at our offices on April 24, 1973. Pending completion of the Regulatory staff review, ~~we request that~~ you should limit operation of Unit No. 2 to a power level no greater than 75% of licensed full power. Further, ~~we request that~~ you should continue to observe the special operating instructions imposed during the latter part of Cycle 1 (enclosed as an attachment to our letter of September 12, 1972).

We have concluded that operation of H. B. Robinson Unit No. 2 in accordance with these restrictions will provide an acceptable margin of safety for Unit No. 2 Cycle 2 operation pending completion of our review. Our intention is to complete this review as soon as possible consistent with our other commitments. In the interim, operation of the plant should be in accord with limitations listed above ~~if you consider it necessary and~~ unless our approval ~~of~~ ~~[illegible] has been~~ of modifications to the limitations has been obtained.

Sincerely,

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

cc: see next page

APR 12 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

By letter dated February 22, 1973, you proposed a change to the Technical Specifications appended to License No. DPR-23 for Unit No. 2 at the H. B. Robinson Steam Electric Plant. The proposed change would apply to Technical Specification 3.10.1.5 and would permit operation at criticality for two hours per year with all full length control rods completely inserted except for the rod with maximum worth. In your application, you state that the purpose of the proposed mode of operation is to increase the accuracy of your estimates of shutdown margin.

We have reviewed your application and we have found that it is incomplete. To complete the application, it will be necessary for you to describe in greater detail the intended measurements, to submit a safety analysis as required by paragraph 50.59(d) of 10 CFR Part 50, to propose changes to the relevant bases in the Technical Specifications, to provide your estimate of the increase in accuracy with which safety margin can be estimated from reactivity measurements with the full length control rods completely inserted, and to discuss the reasons for the increase in accuracy. Please submit the additional information by June 15, 1973.

Please contact us if you have any questions regarding our request for additional information.

Sincerely,

Original signed by
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

LB

cc: see next page

OFFICE ▶						
SURNAME ▶						
DATE ▶						

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
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R. W. Woodruff, L:OR-1 (2)
S. A. Teets, L:OR-1

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SURNAME ▶	RWWoodruff:dc	SATeets		RSchemel		
DATE ▶	3/27/73	3/27/73	3/ /73	4/ /73		

APR 12 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

Paragraph 50.36(c)(2) of 10 CFR Part 50 and the Section 3.0 of the Technical Specifications appended to License No. DPR-23 require that certain actions be taken in the event that a limiting condition for operation of Unit No. 2 at the H. B. Robinson Steam Electric Plant is violated. The action generally required is reactor shutdown. Section 4.0 of the Technical Specifications specifies the minimum time intervals between observations of the limiting conditions for operation, i.e., the minimum time intervals between surveillance tests.

By letter dated February 28, 1973, you stated that certain surveillance tests should not be performed when the reactor is shut down. Further, you requested changes to Section 4.0 which would suspend, when the reactor is shut down, the requirement for all surveillance tests which would require shutdown if the associated limiting conditions for operation were violated. Your letter does not identify the surveillance tests considered and does not include a safety analysis.

Paragraph 50.36(c)(2) and the Technical Specifications 3.3.1.1 and 3.3.1.2 require that the reactor be shut down if the boron injection tank contains less than 900 gallons of water boronated to less than 20,000 ppm. Independent of the operating status of the reactor, Technical Specification 4.1.2 requires periodic surveillance of the boron concentration in the boron injection tank. If we authorized the proposed change, surveillance of the boron concentration clearly would not be required for the cold shutdown condition, and the requirement with regard to the hot shutdown condition would be ambiguous. Because of the results of the safety analysis for the steam line break at hot shutdown as presented in Section 14.2.5 of the Final Safety Analysis Report, clearly the need exists for boron injection capability at hot shutdown. Further, we conclude that it is prudent

LB

to maintain boron injection capability whenever fuel is in the reactor vessel. However, with regard to boron injection capability, the Technical Specifications include no limiting conditions for operation when the reactor is in the hot shutdown, cold shutdown, or refueling shutdown condition. In addition, we note that Technical Specifications 1.2.1, 1.2.2, and 1.2.3 do not include all reactor shutdown conditions.

In order for us to take action on your application for the proposed change to the Technical Specifications, it will be necessary for you to identify the surveillance tests which you believe should not be performed when the reactor is shut down and to provide safety analyses supporting the proposed change. If you wish to pursue the proposed change to the Technical Specifications, please revise your application within sixty days of receipt of this letter. The revised application must identify the surveillance tests in question and include the necessary safety analyses as required by paragraph 50.59(d) of 10 CFR Part 50. Please submit the revised application by June 15, 1973.

In any event, please propose, prior to reactor startup following the current refueling outage, changes to the Technical Specifications which would identify in Section 1.0 all shutdown conditions and which would set forth in Section 3.0 limiting conditions for operation with regard to boron injection capability when the reactor is shut down.

Please contact us if you have any questions regarding these requests.

Sincerely,

Original signed by
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
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RO (3)
OGC
RJSchemel

SEE PREVIOUS YELLOW FOR OTHER CONCURRENCES

OFFICE ▶	L:OR-1					
SURNAME ▶	RJS RJSchemel					
DATE ▶	4/12/73					

to maintain boron injection capability whenever fuel is in the reactor vessel. However, with regard to boron injection capability, the Technical Specifieations include no limiting conditions for operation when the reactor is in the hot shutdown, cold shutdown, or refueling shutdown condition. In addition, we note that Technical Specifications 1.2.1, 1.2.2, and 1.2.3 do not include all reactor shutdown conditions.

In order for us to take action on your application for the proposed change to the Technical Specifications, it will be necessary for you to identify the surveillance tests which you believe should not be performed when the reactor is shut down and to provide safety analyses supporting the proposed change. If you wish to pursue the proposed change to the Technical Specifications, please revise your application within sixty days of receipt of this letter. The revised application must identify the surveillance tests in questions and include the necessary safety analyses as required by Paragraph 50.59(d) of 10 CFR Part 50. ~~If you do not submit the revised application by June 15, 1973, please consider this letter to constitute denial of the proposed change in accordance with Paragraph 2.108 of 10 CFR Part 2.~~

No/

In any event, please propose, prior to reactor startup following the current refueling outage, changes to the Technical Specifications which would identify in Section 1.0 all shutdown conditions and which would set forth in Section 3.0 limiting conditions for operation with regard to boron injection capability when the reactor is shut down.

Letter is ok if this sentence is stricken

Please contact us if you have any questions regarding these requests.

SB ~~Your application~~

Sincerely,

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
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S. A. Teets
T. J. Carter

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SURNAME ▶	RWWoodruff:dc	SATeets		RJSchemel		
DATE ▶	3/29/73	3/30/73	3/ /73	3/ /73		

APR 6 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. N. B. Bessac, Manager
Nuclear Generation
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

Because of indications of degradation of some fuel cladding and steam generator tubes at Robinson-2, we requested that the steam generators be hydrostatically tested each month. By letter dated January 29, 1973, you submitted a description of hydrostatic testing of the Robinson-2 steam generators, and you requested that our requirement for monthly hydrostatic testing be rescinded. Further, by letter dated February 16, 1973, you submitted a report (NSD-E-NLB-672) describing: eddy current testing of some of the steam generator tubes, plugging of defective steam generator tubes, and corrective action to prevent additional degradation of steam generator tubes.

Before we can rescind the requirement for monthly hydrostatic testing, it will be necessary for us to have reasonable assurance that all tubes with significant defects have been plugged and that the corrective action has been effective in arresting degradation. To assist us in this regard, it will be necessary for you to submit for our review a discussion of the results of eddy current testing being performed during the current outage, including a comparison of these results to results obtained previously, and to submit copies of WCAP-7452 which discusses the use of phosphates to control steam generator corrosion and which is referenced in NSD-E-NLB-672.

Forty copies of your submittal will be needed, and if any part of it is proprietary, it will be necessary for you to observe the requirements of Section 2.790 of 10 CFR Part 2. If you have any questions regarding our requirements, please do not hesitate to call Mr. Roger Woodruff.

Sincerely,

ABurger

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

LB

cc: See next page

OFFICE ▶						
SURNAME ▶						
DATE ▶						

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
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SURNAME ▶	RWoodruff:lb	SATeets	RJSchemel			
DATE ▶	4/5/73	4/5/73	4/6/73			

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APR 6 1973

L:MPP:PKM
50-261

Carolina Power and Light Company
Attn: Mr. E. E. Utley, Vice President
Bulk Power Supply
P. O. Box 1551
Raleigh, North Carolina 27602

Gentlemen:

In response to your letter of March 6 concerning instructions for reporting transfers on Form AEC-741, we are considering amendments to the instructions (a) to allow the receiver who accepts shipper's values additional time after receipt of nuclear material to complete and distribute the form, and (b) to permit acceptance of shipper's corrected values without a referee determination in those cases where the receiver has not made independent measurements.

Thank you for bringing these matters to our attention.

Sincerely,

Original Signed by
Ralph G. Page

R. G. Page, Chief
Materials and Plant Protection
Branch
Directorate of Licensing

OFFICE ▸	L:MPP	L:MPP				
SURNAME ▸	PKMorrow:ra	RGPage				
DATE ▸	4/6/73	4/6/73				

Form AEC-318 (Rev. 9-53) AECM 0240

Carolina Power & Light Company

March 6, 1973

Mr. R. G. Page, Chief
Materials and Plant Protection Branch
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

REVISED INSTRUCTIONS FOR REPORTING NUCLEAR MATERIAL TRANSFERS ON FORM AEC - 741 - NUCLEAR MATERIAL TRANSFER REPORT

Dear Mr. Page:

Revised instructions for reporting receipt of special nuclear material impose requirements that present serious problems for Carolina Power & Light Company. The receiver of reportable quantities of special nuclear material is now required to acknowledge and sign form AEC-741 on the day of receipt of the material if the receiver plans to accept the shipper's measurement data without making independent measurements.

Fuel assemblies shipped to our H. B. Robinson and Brunswick Nuclear Plant sites are received on the same day as shipped but the transfer reports are mailed by the shipper and are not received until several days later. The reference to independent measurements implies that the instructions are directed toward shipment of bulk SNM such as UF_6 and not toward shipment of fuel assemblies.

The revised instructions also appear to be directed toward shipment of bulk material with respect to acknowledgement of "Corrected Copy" AEC-741's. The revised instructions prohibit accepting shipper's adjustment without independent measurement by referee. The instructions should provide a mechanism for accepting fuel manufacturer's revised assembly content values.

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

For these reasons Carolina Power & Light requests that you reconsider the new requirements in their application to fuel assembly shipments. We would also appreciate an opportunity to comment on such changes in requirements prior to their becoming effective.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

LHM:lw

cc: Messrs. H. R. Banks
L. H. Martin
L. E. Smith
R. A. Watson

MAR 29 1973

Docket No. 50-261

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

Gentlemen:

In our letter dated December 15, 1972, we requested: (a) that you compare, for Unit No. 2 at the H. B. Robinson Steam Electric Plant, the design of high energy lines outside containment to the requirements of General Design Criterion No. 4, (b) that you provide analyses and other relevant information needed to determine the consequences of failure of these high energy lines, and (c) that you identify modifications, if any, needed to protect structures, systems, or components important to safety.

By letter dated February 6, 1973, you responded by providing information with regard to the routing of main steam lines outside containment and by providing conclusions with regard to the consequences of failure of these lines. Your response did not include analyses supporting each of the conclusions and did not address the main feedwater lines and small diameter high energy lines. During a meeting in Bethesda on January 17, 1973, you were informed that high energy lines having smaller diameters than the main steam and feedwater lines must also be included in your evaluation. Within 60 days of receipt of this letter, please submit an evaluation of the main feedwater lines and the small diameter lines and submit analyses which support your conclusions regarding the main steam, main feedwater and small diameter lines.

If you have questions with regard to our request, please do not hesitate to call us.

Sincerely,

Original Signed by:
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

8-9

cc: OFFICE ▶	See next page					
SURNAME ▶						
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cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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FEB 27 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

By letter dated September 28, 1972, you provided the additional information which we requested regarding an abnormal occurrence at Robinson-2 involving dilution of boron in the boron injection tank, and in addition, you requested clarification of the Technical Specifications pertaining to the length of time that the concentration of boron in the boron injection tank may be less than the concentration required for operation (limiting condition for operation) of the reactor.

Paragraph 50.36(c)(2) of 10 CFR Part 50 requires, when a limiting condition for operation is not met, that the licensee shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. The Technical Specifications for Robinson-2 do not permit, in the event of low boron concentration in the boron injection tank, any remedial action other than shutting down the reactor. Therefore, you are required in that event to initiate immediately routine operating procedures for putting the reactor in the hot shutdown condition, and if the limiting operating condition is not met when the hot shutdown condition is established, you are then required to initiate immediately routine operating procedures for putting the reactor in the cold shutdown condition.

By letters dated November 10 and 29, 1972, you provided a report of another abnormal occurrence involving dilution of boron in the boron injection tank and efforts taken to assure that temperature everywhere in the boron injection system is above the boron saturation temperature. Based on the information presented, we conclude that your action in this regard is significant. Nevertheless, because the possibility of boron dilution remains, due to possible failure of valves to seat for other reasons, we find that additional surveillance of the boron concentration in the boron injection tank, is warranted. Therefore, please submit proposed changes to Technical Specification 4.1.2 which would increase the frequency of surveillance of the boron concentration in the boron injection tank. The frequency of surveillance proposed should be commensurate with the importance

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FEB 27 1973

of boron injection as the backup system for shutting the reactor down and should be based on the need for providing the operator with information that will lead to corrective action before a limiting condition for operation is violated.

Please respond to this request within 10 days of receipt of this letter. If you wish to discuss your response with us, please call Mr. Roger Woodruff.

Sincerely,

Original Signed by:
Robert J. Schemel

for Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Memo to File (Safety Evaluation)

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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FEB 27 1973

Files (Robinson-2, Docket No. 50-261)
THRU: R. J. Schemel, Chief, ORB #1, L

SAFETY EVALUATION RE BORON SURVEILLANCE

Safety reviews by the licensee (See letters from Carolina Power & Light Company dated September 28, November 10 and 29, 1972) indicate that dilution of the boron in the boron injection tank was caused by precipitation of boric acid crystals which resulted in failure of valves to seat and in plugging of a line. The licensee has identified heat sinks in valves and lines and has added heaters and thermal insulation to maintain temperatures above the precipitation temperature for boric acid. In addition, administrative requirements have been established by the licensee for periodic recirculation of boric acid solution to prevent local concentration and precipitation.

While these physical and administrative changes are good, the possibility of boron dilution remains because of the possibility of failure to eliminate all heat sinks and failure of valves to seat for other reasons. In view of the abnormal occurrences which have occurred in the past and may occur in the future and in view of the importance of the boron injection system as a backup system for shutting the reactor down, monthly surveillance of boron concentration is simply not adequate. Therefore, we are sending to CP&L a request that they propose more frequent surveillance of the boron concentration in the boron injection tank.

S/Roger W. Woodruff
Operating Reactors Branch #1
Directorate of Licensing

FEB 22 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Manager
Bulk Power Supply
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

At the time Unit No. 2 at the H. B. Robinson Steam Electric Plant was being licensed, the Commission was developing requirements which were subsequently set forth in Section 50.34(b)(6)(ii) and Appendix B of 10 CFR Part 50. These requirements identify managerial and administrative controls to be used by licensees to assure safe operation of nuclear power plants. To provide guidelines for licensees in establishing adequate programs in this regard, Safety Guide No. 33, Quality Assurance Program Requirements (Operation), was issued on November 3, 1972.

In order that we can determine the extent to which your quality assurance program meets the requirements of Section 50.34(b)(6)(ii) and Appendix B, please submit a description of your program. In addition, please compare your program to the guidelines expressed in Safety Guide No. 33. In your submittal, identify any areas in which the program does not meet the guidelines in Safety Guide No. 33 and the extent to which the guidelines are not met.

The information which we are requesting should be submitted within ninety days of receipt of this letter.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

cc: see next page

RG

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cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
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R. W. Woodruff, L:OR-1 (2)
S. A. Teets, L:OR-1

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DATE ▸	2/22/73	2/22/73	2/22/73	2/22/73		

Docket

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-261

January 16, 1973

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

Gentlemen:

Enclosed is the errata sheet for the modifications to the "General Information Guide for Consideration of the Effects of a Piping System Break Outside Containment" which was transmitted to you by letter dated December 18, 1972.

Sincerely,

Donald J. Skovholt
Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Errata Sheet

cc w/enclosure:
George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

ERRATA SHEET FOR "GENERAL INFORMATION REQUIRED FOR CONSIDERATION OF THE EFFECTS OF A PIPING SYSTEM BREAK OUTSIDE CONTAINMENT"

The following lists the changes that have evolved on our initial information request:

1. Page 2, Item 2--Insert the following in 2. to precede the existing first sentence:

   "Design basis break locations should be selected in accordance with the following pipe whip protection criteria; however, where pipes carrying high energy fluid are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width."

2. Page 2, Item 2(a)(2)--Change nomenclature to read "any intermediate locations between terminal ends where the primary plus secondary stress intensities $S_{\underline{\underline{n}}}$...

3. Page 4, Item 2.(b)(2)--Change 0.9 $(S_h + S_A)$ to 0.8 $(S_h + S_A)$.

4. Page 6, Item 7--Add "structural" to read "The structural design loads..."

5. Page 7, Item 11.(a)--Add "required" so as to read, "Loss of required redundancy..."

6. Page 7, Item 11.(a)--Delete "the steam line break" and replace with "that" to read "...the consequences of that accident..."

7. Page 8, Item 11.(b)--Replace (b) with the following: (b) "Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted but a loss of function is not permitted. For such situations plant shutdown is required."

8. Page 8, Item 13--Change wording in the first sentence to read "Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy fluid line break."

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Scinto, OGC

Docket No. 50-261

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

Gentlemen:

By letter dated September 27, 1972, you proposed a change to the Technical Specifications pertaining to fuel handling for the H. B. Robinson Unit No. 2.

During our review of the proposed change, we found that additional information is required in order to complete our evaluation, and we discussed this requirement with members of your staff. Please provide the information described in the enclosure to this letter in three signed originals and thirty-seven additional copies.

Sincerely,

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Enclosure:
Request for Additional Information

cc w/enclosure:
George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

~~See previous yellow for other concurrences.~~

OFFICE ►	L:OR-1	L:OR-1	L:OR-1			
SURNAME ►	VLRooney:dc	SATeets	RJSchemel			
DATE ►	12/19/72	12/20/72	12/20/72			

Form AEC-318 (Rev. 9-53) AECM 0240

REQUEST FOR ADDITIONAL INFORMATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

Additional information regarding the evaluation of fuel handling accidents is needed as follows:

1. The Robinson safety evaluation of fuel handling accidents appears to assume that activity released into the upper half of the containment vessel:

   a. undergoes perfect mixing with the atmosphere in the upper half of the containment vessel, and

   b. is released to the atmosphere in a quantity determined by one containment exhaust fan operating at a flow rate of 35,000 cfm for five minutes.

   The first assumption appears nonconservative because 8000 cfm of the 35,000 cfm exhaust air flow is drawn directly from above the refueling water surface before mixing can occur. The second assumption seems conservative, but it is not clear that the two assumptions taken together yield conservative results. Please justify the conservatism of the results, or, if this is not possible, provide a new evaluation which you show to be conservative.

2. You gave certain assumptions in your evaluation, and then stated that all other assumptions were made in accordance with Safety Guide 25. In order to meet a variety of conditions, Safety Guide 25 permits various assumptions, and in order to complete our evaluation, we must know which of the following assumptions you chose.

   a. Was maximum fuel rod pressurization assumed less than 1200 psig?

   b. Was minimum water depth taken to be 23 feet?

   c. Was ground level release assumed?

   d. What building wake factor was used?

3. Do the core inventories given in Table 1 apply at the time of shutdown, or after the 100-hour decay period?

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4. Furnish a description of the spent fuel building ventilation system as modified by the charcoal filter installation. It is suggested that the description be provided in the form of revised pages to the FSAR in order to accomplish the dual purpose of meeting the information needs for the safety evaluation and the update of the FSAR with respect to the planned equipment change.

5. Propose Technical Specifications which establish performance levels for the ventilation system, including filter performance.

6. Propose Technical Specifications which establish surveillance requirements for the ventilation system and filter.

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Docket No. 50-261

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

Gentlemen:

The Regulatory staff's continuing review of reactor power plant safety indicates that the consequences of postulated pipe failures outside of the containment structure, including the rupture of a main steam or feedwater line, need to be adequately analyzed and documented by licensees and applicants, and evaluated by the staff as soon as possible. Criterion No. 4 of the Commission's General Design Criteria, listed in Appendix A of 10 CFR Part 50, requires that:

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

The previous version of the Commission's General Design Criteria also reflects the above requirements.

Thus, a nuclear plant should be designed so that the reactor can be shut down and maintained in a safe shutdown condition in the event of a postulated rupture, outside containment, of a pipe containing a high energy fluid, including the double-ended rupture of the largest pipe in the main steam and feedwater systems. Plant structures, systems, and components important to safety should be designed and located in the facility to accommodate the effects of such a postulated pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

We understand that no steam lines traverse the Auxiliary and Intermediate Buildings and that rupture of these lines would not jeopardize safety related equipment; however, to confirm our understanding of this situation, please submit to us a description of the routing of steam and feedwater lines. In addition, we request that you provide us with analyses and other relevant

information needed to determine the consequences of such an event, using the guidance provided in the enclosed general information request. The enclosure represents our basic information requirements for plants now being constructed or operating. You should determine the applicability, for H. B. Robinson 2, of the items listed in the enclosure.

If the results of your analyses indicate that changes in the design of structures, systems, or components are necessary to assure safe reactor shutdown in the event this postulated accident situation should occur, please provide information on your plans to revise the design of your facility to accommodate the postulated failures described above. Any design modifications proposed should include appropriate consideration of the guidelines and requests for information in the enclosure.

We will also need, as soon as possible, estimates of the schedule for design, fabrication, and installation of any modifications found to be necessary. Please inform us within 7 days after receipt of this letter when we may expect to receive an amendment with your analysis of this postulated accident situation for H. B. Robinson 2, a description of any proposed modifications, and the schedule estimates described above. Sixty copies of the amendment should be provided.

A copy of the Commission's press announcement on this matter is also enclosed for your information.

Sincerely,

Original signed by:
Roger S. Boyd

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

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Enclosures:
As stated

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:
   (a) Both of the following piping system conditions are met:
      (1) the service temperature is less than 200° F; and
      (2) the design pressure is 275 psig or less; or
   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or
   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or

(d) The internal energy level[1] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

2. The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

   (a) ASME Section III Code Class I piping[2] breaks should be postulated to occur at the following locations in each piping run[3] or branch run:

      (1) the terminal ends;

      (2) any intermediate locations between terminal ends where the primary plus secondary stress intensities S_m (circumferential or longitudinal) derived on an elastically

---

[1]The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

[2]Piping is a pressure retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

[3]A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4] exceeds 2.0 S_m[5] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) any intermediate locations between terminal ends where the cumulative usage factor (U)[6] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) at intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) the terminal ends;

---

[4] Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

[5] S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

[6] U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed $0.9\ (S_h + S_A)$[7] or the expansion stresses exceed $0.8\ S_A$; and

(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under 2 above should be equivalent to the following:

(a) Longitudinal[8] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

---

[7] S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

[8] Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

(b) Circumferential[9] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:

(a) The locations and number of design basis breaks on which the dynamic analyses are based.

(b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

(c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.

(d) Diagrams of mathematical models used for the dynamic analysis.

(e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structure, systems, or components important to safety, such as the control room.

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[9] Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet and reactive forces including:

   (a) Pipe restraint design to prevent pipe whip impact;

   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;

   (c) Separation of redundant features;

   (d) Provisions to separate physically piping and other components of redundant features; and

   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:

   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;

   (b) The allowable design stresses and/or strains; and

   (c) The load factors and the load combinations.

7. The design loads, including the pressure and temperature transients, the dead, live and equipment loads; and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including nonseismic Category I structures, caused by the accident, will not cause failure of any other structure in a manner to adversely affect:
    (a) Mitigation of the consequences of the accidents; and
    (b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:
    (a) Loss of redundancy in any portion of the protection system (as defined in IEEE-279), Class IE electric system (as defined in IEEE-308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of the steam line break accident and place the reactor(s) in a cold shutdown condition; or

(b) Loss of the ability to cope with accidents due to ruptures of pipes other than a steam line, such as the rupture of pipes causing a steam or water leak too small to cause a reactor accident but large enough to cause electrical failure.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown of the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a steam line or feedwater line break. The information required for our review should include the following:

    (a) Identification of all electrical equipment necessary to meet requirements of 11 above. The time after the accident in which they are required to operate should be given.

    (b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

    (c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingment from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

(d) An evaluation of the capability for safety related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

AEC

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. P-429
Contact: Frank Ingram
Tel. 301/973-7771

FOR IMMEDIATE RELEASE
(Wednesday, December 13, 1972)

AEC REGULATORY STAFF REQUESTS DATA ON PIPE BREAKS IN NUCLEAR PLANTS

The Atomic Energy Commission's Regulatory Staff is asking all utilities that operate nuclear power plants or have applied for operating licenses to assess the effects on essential auxiliary systems of a major break of the largest main steam or feedwater line. These lines carry steam from inside the reactor containment building to the main turbine in the turbine building, and hot feedwater back from the turbine condenser. The utility assessments will be evaluated by the AEC's Regulatory Staff.

The probability of a steam-line rupture is low. Nonetheless it will have to be considered in the AEC's safety evaluation.

The review of the pipe break problem has been under way for several weeks. It was started after the Advisory Committee on Reactor Safeguards received a letter raising questions about the location of pipes in the two-unit Prairie Island plant in Minnesota.

The Regulatory Staff has reviewed the Northern States Power Company application to operate Prairie Island, and on the basis of data available it has concluded that design changes will be required at Prairie Island.

Based on the new information--to be submitted by utilities as soon as possible--the Staff will determine what corrective action, if any, is necessary in each case. The changes could include such steps as relocating piping, providing venting of compartments, the addition of piping restraints, and, in some cases, structural strengthening.

#

NOV 24 1972

Docket No. 50-261

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

Gentlemen:

Because of indications at H. B. Robinson Unit No. 2 of degradation of some fuel cladding and steam generator tubes, we requested by letter dated October 5, 1972, that you submit by October 24, 1972, later extended to October 31, 1972, a proposed change to the Technical Specifications, which would require additional surveillance of the steam generators to assure that they would retain fission products in the event of any of the postulated accidents normally addressed in Safety Analysis Reports. The proposed change was to include appropriate bases and supporting information for the bases. In addition, we requested that you be prepared to perform the first such surveillance test by October 31, 1972, and submit revised accident analyses by December 31, 1972.

In your letter dated October 27, 1972, you suggested that the existing limitations on primary to secondary leakage provides assurance that flawed steam generator tubes will not fail as a result of a steam-line-break accident. We have performed a preliminary review of the information attached to your letter and we find that it does not provide sufficient information for us to conclude that all flaws which can lead to failure of tubes during a steam-line-break accident can be detected by this technique.

Because the power transient resulting from a major steam-line-break accident may lead to release of a substantial fraction of the fission products in the cladding gaps of degraded fuel rods and because we do not now have adequate assurance that the steam generator tubes would retain these fission products, we are hereby requesting that CP&L perform a pressure test on each of the steam generators to assure that they are at present capable of withstanding the pressure transient which would be caused by a major steam-line-break accident. In this

regard, we consider either of the following sets of test conditions to be adequate:

(a) 1900 psi pressure differential across the tube walls with the tube walls at 400°F, or

(b) 2300 psi pressure differential across the tube walls with the tube walls at the cold shutdown temperature.

Please perform the first such test within one week of the date of this letter and monthly thereafter.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso
Deputy Director for
Reactor Projects
Directorate of Licensing

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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DATE ▶	11/20/72	11/20/72	11/20/72	11/20/72	11/20/72	11/24/72

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R. W. Woodruff, L:OR-1
S. A. Teets, L:OR-1

DEC 12 1972

Docket No. 50-261

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

Gentlemen:

In the future, please provide forty copies of all items submitted other than annual financial reports and the items that have copy requirements specified in Section 50.30 and Appendix D of 10 CFR Part 50.

Sincerely,

/s/

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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SURNAME ▸	SATeets:dc	RWWoodruff	RJSchemel	DJSkovholt		
DATE ▸	12/7/72	12/11/72	12/12/72	12/12/72		

Form AEC-318 (Rev. 9-53) AECM 0240

Docket File

NOV 20 1972

Docket No. 50-261

Carolina Power & Light Company
ATTN: P. S. Colby
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

RE: H. B. ROBINSON STATION UNIT NO. 2

Gentlemen:

The Commission's Regulatory Staff has completed a review of fuel densification and its effect on reactor operation including transients and postulated loss-of-coolant accidents. The Staff's investigations and conclusions are reported in "Technical Report on Densification of Light Water Reactor Fuels" dated November 14, 1972, a copy of which is enclosed for your information and guidance. This report concludes that densification of fuel may occur and that the resulting formation of fuel column gaps should be anticipated in all light water reactor fuels. The report also provides the essential elements to be included in calculational models used to account for the effects of fuel densification.

The Regulatory Staff believes that the fuel in the subject facility ~~is susceptible to~~ densification. Therefore, we request that you provide the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance provided in the enclosed report. These analyses should be provided for your present and next fuel loadings. If the analyses indicate that changes in design or operating conditions are necessary to maintain required margins, you should submit proposed changes and operating limitations with the analyses.

DL

NOV 20 1972

In order that the Regulatory Staff can conduct an expeditious and orderly review of these matters, we request that you submit the analyses and additional information within 45 days from the date of this letter. It is requested that this information be provided with one signed original and thirty-nine additional copies.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
Technical Report on Densification
(November 14, 1972)

cc w/o enclosure:
George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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RTedesco, L:CS (2)
SDMacKay, L:ORB #1

This letter does not acknowledge our best estimate of the present condition of Robinson fuel and does not recognize the potential consequences of a steam-line-break accident. RWW 11/20/72

OFFICE ▸	L:ORB #1	L:ORB #1	L:ORB #1	L:OR	L:RP	
SURNAME ▸	SDM SDMacKay:sjh	RWWoodruff	RJS RJSchemel	DJSkovholt	AGiambusso	
DATE ▸	11/20/72	11/ /72	11/20/72	11/20/72	11/21/72	

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J. McGough

OCT 3 - 1972

Docket No. 50-261

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

Gentlemen:

By letters dated September 29, 1971, and September 8, 1972, you proposed a reduction in the nuclear hot channel factor contained in the Technical Specifications for the H. B. Robinson Unit No. 2.

During our review of the proposed change, we found that additional information was required in order to complete our evaluation of operation with the reduced nuclear hot channel factor and the present control rod insertion limits. We have discussed this matter with members of your staff on several occasions since June of this year. Please provide the information described in the enclosure to this letter by November 1, 1972, in three signed originals and thirty-seven additional copies.

Sincerely,

/s/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

D-10/5/72

OFFICE ▶	L:OR-1	L	L:OR-1	L:OR		
SURNAME ▶	J.McGough RWoodruff	SATeets	RJSchemel	DJSkovholt		
DATE ▶	9/29/72	10/2/72	10/3/72	10/3/72		

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 3, 1972

Docket No. 50-261

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

Gentlemen:

By letters dated September 29, 1971, and September 8, 1972, you proposed a reduction in the nuclear hot channel factor contained in the Technical Specifications for the H. B. Robinson Unit No. 2.

During our review of the proposed change, we found that additional information was required in order to complete our evaluation of operation with the reduced nuclear hot channel factor and the present control rod insertion limits. We have discussed this matter with members of your staff on several occasions since June of this year. Please provide the information described in the enclosure to this letter by November 1, 1972, in three signed originals and thirty-seven additional copies.

Sincerely,

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

REQUEST FOR ADDITIONAL INFORMATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

Presently, H. B. Robinson is authorized operation with a total nuclear hot channel factor of:

$F_q^N = 3.13$

and control rod insertion limits as specified in Figure 3.10-1 of the Technical Specifications. Your proposal for reactor operation with a lower overall nuclear hot channel factor may require new control rod insertion limits in order to limit the maximum fuel power density to 15.8 kW/ft at 2200 MWt. In order for us to evaluate your request, please provide us with the following information:

1. Using the worst case rod configurations permissible under the presently allowable rod insertion limits, provide a summary analysis of the predicted maximum total hot channel factors and associated fuel power densities expected during steady state, normal and anticipated operational transients. Include in your study, the effects of core cyclic changes, fuel reloads, unrestricted part length rod insertion and any other significant parameters that may influence the conclusions. Provide a list of the assumptions used for each case examined.

2. Describe the method and frequency of any surveillance measures that will be implemented to assure compliance with the proposed reduced total nuclear hot channel factor.

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R. R. Maccary
R. Mattson

Docket No. 50-261

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

Gentlemen:

As we discussed with your staff on August 18, 1972, the Final Safety Analysis Report for H. B. Robinson Unit No. 2 includes an analysis of the steam-line-break accident in which it is tacitly assumed that the first and second barriers to the release of fission products would not fail if the accident were to occur. Because of cracking of some steam generator tubes and because of indicated degradation of some of the fuel cladding, this assumption may no longer be warranted. Since the consequences of an accident previously analyzed may be increased, an unreviewed safety question exists. Please provide a revised analysis of the steam-line-break accident and any other accident previously analyzed which may be affected. The analysis of the steam-line-break accident should include the considerations described in the enclosure to this letter.

Until our review of your analysis is completed, please demonstrate periodically by surveillance testing the capability of the steam generator tubes to withstand the pressure transient that would result from the steam-line-break accident. A hydraulic pressure test would be satisfactory provided that the test pressure is great enough to allow for: (a) the maximum pressure differential across the tubes resulting from failure of the largest steam line between containment and the isolation valves, (b) the effect of rapid application of pressure in the accident case as compared to slow application in the test case, (c) the effect of temperature for the accident case as compared to actual temperature in the test case, and (d) margin of safety.

Please submit by October 24, 1972, a proposed change to the Technical Specifications which would specify the test conditions for surveillance of the steam generators, provide bases for the test conditions, and provide supporting

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information for the bases; and be prepared by October 31, 1972, to perform the first such surveillance test. Please submit the revised analysis of the steam-line-break accident by December 31, 1972. If we can be of assistance in these regards, please call Mr. Roger W. Woodruff.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

OFFICE ▶	L:OR-1	L	L:OR-1	L	L	L:OR
SURNAME ▶	RWWoodruff:pdl	SATeets	RJSchemel	DKnuth	RRMacCary	DJSkovholt
DATE ▶	9/25/72	9/25/72	9/25/72	10/3/72	10/4/72	10/5/72 Signed

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 5, 1972

Docket No. 50-261

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

Gentlemen:

As we discussed with your staff on August 18, 1972, the Final Safety Analysis Report for H. B. Robinson Unit No. 2 includes an analysis of the steam-line-break accident in which it is tacitly assumed that the first and second barriers to the release of fission products would not fail if the accident were to occur. Because of cracking of some steam generator tubes and because of indicated degradation of some of the fuel cladding, this assumption may no longer be warranted. Since the consequences of an accident previously analyzed may be increased, an unreviewed safety question exists. Please provide a revised analysis of the steam-line-break accident and any other accident previously analyzed which may be affected. The analysis of the steam-line-break accident should include the considerations described in the enclosure to this letter.

Until our review of your analysis is completed, please demonstrate periodically by surveillance testing the capability of the steam generator tubes to withstand the pressure transient that would result from the steam-line-break accident. A hydraulic pressure test would be satisfactory provided that the test pressure is great enough to allow for: (a) the maximum pressure differential across the tubes resulting from failure of the largest steam line between containment and the isolation valves, (b) the effect of rapid application of pressure in the accident case as compared to slow application in the test case, (c) the effect of temperature for the accident case as compared to actual temperature in the test case, and (d) margin of safety.

Please submit by October 24, 1972, a proposed change to the Technical Specifications which would specify the test conditions for surveillance of the steam generators, provide bases for the test conditions, and provide supporting

information for the bases; and be prepared by October 31, 1972, to perform the first such surveillance test. Please submit the revised analysis of the steam-line-break accident by December 31, 1972. If we can be of assistance in these regards, please call Mr. Roger W. Woodruff.

Sincerely,

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

REQUEST FOR ADDITIONAL INFORMATION AND ANALYSIS

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

A. Additional information pertaining to the capability of the fuel rods and the steam generators to retain fission products is needed as follows:

   1. With regard to the steam generators, provide:

      (a) A description of their present condition,

      (b) Assurance that all cracked tubes have been found,

      (c) Assurance that the action taken to arrest degradation of the tubes is effective, and

      (d) The critical flaw size for the tubes as a function of pressure and temperature.

   2. Provide a quantitative description of fuel pellet dishing and a quantitative discussion of the design criteria on which dishing is based.

B. A revised analysis of the consequences of double ended failure of the largest steam pipe between containment and the isolation valves is needed which considers the possible effect of the failure on fuel clad and steam generator tubes. The revised analysis should include:

   1. The assumption that the trip points are set at the most unconservative limits permitted by the Technical Specifications,

   2. A calculation, as a function of initial power, of the maximum stress and strain in fuel clad, assuming that the clad is initially crimped so that it bears on the top of the fuel column,

   3. An estimate of the stress and strain at which fuel clad failure would occur,

4. A calculation of the maximum stress in the steam generator tubes as a result of rapid pressure transients, and

5. An estimate of the stress at which steam generator tube failure would occur.

C. If there is any uncertainty with regard to the capability of fuel clad and a steam generator tube to withstand the transients resulting from the steam line break, submit a calculation of the dose at the site boundary.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 5 - 1972

Files (H. B. Robinson-2, Docket No. 50-261)
THRU: R. C. Schemel, Chief, ORB #1, L

UNREVIEWED SAFETY QUESTION RE BARRIERS TO RELEASE OF FISSION PRODUCTS

SUMMARY

Previous analyses of the steam-line-break accident assume that other barriers to the release of fission products do not fail during the course of the accident. Degradation of the Robinson-2 steam generator tubes and fuel clad is indicated to the extent that we are uncertain of their capability to withstand the pressure and power transients resulting from a steam-line-break accident at power. If these barriers should fail, the potential doses at the site boundary would be greater than those previously analyzed.

Additional analysis and information are needed from the licensee regarding the present condition of the steam generators, the present capability of the steam generator tubes, and the capability of crimped fuel clad to survive transients resulting from a steam line break. We conclude that additional surveillance of the steam generators is necessary prior to completion of our review of this safety question.

DISCUSSION

Tube cracking has occurred in the Robinson-2 steam generators due to poor water chemistry on the secondary or shell side. CP&L has taken action to correct the water chemistry and to arrest further tube degradation by adjusting the steam generator recirculation ratio, providing continuous steam generator blowdown, and increasing the phosphate concentration in the secondary coolant(1). The effectiveness of this action is at present unknown to us.

During the May 1972 outage, 69% of the steam generator tubes were inspected. CP&L believes that the 31% not inspected are in good condition because of their location in the steam generators. Thirty tubes known to have cracks extending more than half way through the wall have been plugged(1). Following plugging, the steam generators were hydraulically tested by applying 2250 psi across the tubes. The test revealed no leakage(3).

The cracks in the two sections of damaged tubes removed from the steam generators are longitudinal, 0.5 to 0.6 inch long, and in some cases extend completely through the 40 mil wall. Minor circumferential cracks extend from the longitudinal cracks. Cracked tube sections were tested by slowly applying pressure to 2000 psig at 550°F. Cracks opened and closed as pressure was applied and removed(1). Although no crack growth was noted, we have at present no assurance that cracks are not continuing to grow in some inspected and uninspected tubes in the steam generators.

For a steam-line-break accident at power, the differential pressure across the tubes would increase in roughly 10 seconds(4) from a nominal value of 1465 psi(5,6) to perhaps 1800 psi(1). We have no assurance that a severely cracked tube would not fail during this small but rapid increase in differential pressure.

Axial flux traverses in the Robinson-2 core indicate that fuel clad collapse as experienced in the Ginna core is in progress(2). Once collapsed clad comes in contact with the top of the fuel column, there will remain no free space to accommodate differential thermal expansion between the uranium oxide fuel and the zircaloy 4 clad. For the steam-line-break accident at power, power increases to 109% of full power before scram occurs(7,8). While power and fuel temperature increase, primary coolant temperature decreases, thus the average clad temperature tends to remain constant(4). The transient stress in fuel clad due to increasing fuel temperature is reduced by the dished design(9) of the fuel pellets. The design criteria which resulted in the dish dimensions are unknown to us at this time.

The clad at the top of the fuel column would be stressed beyond the yield point if clad collapse had occurred. A transient thermal stress due to the steam-line-break accident would be superposed on this stress because of thermal expansion of the fuel pellet column. The transient stress is determined by the average temperature of fuel at some point(9) beneath the surface. Assuming that the reactor is operating at 90% of full power and that a steam-line-break accident occurs, the average fuel temperature at this point beneath the fuel surface is very roughly estimated to increase from 1100°F to 1170°F at scram(10). This increase in temperature would cause the fuel column to increase in length by 0.04%(11). At the maximum average clad temperature, 715°F(10), the modulus of elasticity of the clad is 10.8×10^6 psi(12). For 0.04% strain, the transient clad stress would be 4000 psi which would be superposed on an unknown stress above the yield point, ~ 15,000 psi(14), whereas the tensile strength is 18,000 psi at

this temperature and at rapid strain rates[13]. Thus, there may be no margin to failure.

If a steam line break occurs outside containment and upstream from the isolation valve and if the resulting transient fails unpressurized fuel and a steam generator tube, the potential doses at the site boundary would be greater than those previously calculated.

CONCLUSIONS

We conclude that an unreviewed safety question exists for Robinson-2 and that CP&L should be required to submit additional information and analysis of the capability of the fuel rods and the steam generators to provide barriers to the release of fission products in the event of a steam-line-break accident or any other accident previously analyzed.

With regard to the steam-line-break accident, the additional information should include the following:

1. For the steam generators:

    (a) A description of their present condition,

    (b) Assurance that all cracked tubes have been found,

    (c) Assurance that the action taken to arrest degradation of the tubes is effective, and

    (d) The critical flaw size for the tubes as a function of pressure and temperature.

2. A quantitative description of fuel pellet dishing and a quantitative discussion of the design criteria on which dishing is based.

CP&L should submit a revised analysis of the consequences of double ended failure of the largest steam pipe between containment and the isolation valves and should consider the possible effect of the failure on fuel clad and steam generator tubes. The revised analysis should include:

1. The assumption that the trip points are set at the most unconservative limits permitted by the Technical Specifications,

2. A calculation, as a function of initial power, of the maximum stress and strain in fuel clad, assuming that the clad is initially crimped so that it bears on the top of the fuel column,

3. An estimate of the stress and strain at which fuel clad failure would occur,

4. A calculation of the maximum stress in the steam generator tubes as a result of rapid pressure transients, and

5. An estimate of the stress at which steam generator tube failure would occur.

If there is an uncertainty with regard to the capability of fuel clad and a steam generator tube to withstand the transients resulting from the steam line break, CP&L should submit a calculation of the dose at the site boundary.

Until review of this safety question is completed, CP&L should demonstrate periodically by surveillance testing the capability of the steam generator tubes to withstand the pressure transient that would result from the steam-line-break accident. A hydraulic pressure test would be satisfactory provided that the test pressure is great enough to allow for: (a) the maximum pressure differential expected across the tubes during the accident, (b) the effect of rapid application of pressure in the accident case as compared to slow application in the test case, (c) the effect of temperature in the accident case as compared to actual temperature in the test case, and (d) margin.

R. W. Woodruff

R. W. Woodruff
Operating Reactors Branch #1
Directorate of Licensing

Enclosures:
References

cc: A. Giambusso
D. J. Skovholt
T. J. Carter
R. J. Schemel
RO (3)
R. R. Maccary
R. W. Woodruff (2)
S. A. Teets
R. Mattson
M. Jinks (2)
D. Knuth

REFERENCES

1. R. W. Woodruff, letter to Files, Docket No. 50-261, September 1, 1972.

2. W. C. Seidle, letter to J. G. Keppler, August 22, 1972.

3. N. B. Bessac, telecon with R. W. Woodruff, August 18, 1972.

4. Carolina Power & Light Company, "Final Facility Description and Safety Analysis Report, H. B. Robinson Unit No. 2", Docketed November 26, 1968, Figure 14.2.5-3.

5. Ibid., Table 4.1-4.

6. Ibid., Table 4.1-2.

7. Ibid., p. 14.2.5-6.

8. License No. DPR-23, Technical Specification 2.3.1.2(a).

9. N. B. Bessac, telecon with R. W. Woodruff, September 1, 1972.

10. CP&L, op. cit., Table 3.2.2-1.

11. H. C. Brassfield, et. al., GEMP-482, April 1968, p. 69.

12. Ibid., p. 89.

13. Ibid., p. 80.

14. AMAX Specialty Metals Division, "Engineering Information and Data, Zirconium and Hafnium", 10M-101570, p. 23.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket

September 28, 1972

Docket No. 50-261

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

Gentlemen:

As the number of nuclear plants in operation becomes progressively larger, the data from these plants become increasingly important. Therefore, it is highly desirable that data reporting be consistent from plant to plant. This consistency will facilitate the analysis, evaluation and comparison of data on the Safety and Environmental Effects of the operating plants.

The Administrative Controls Section of Technical Specifications delineates reporting requirements. In the near future, we propose to revise this Section of Technical Specifications to assure reporting consistency and to conform as nearly as possible to Safety Guides 16 and 21. We are enclosing copies of these Guides for your information. Also enclosed are suggested Technical Specifications that cover the Plant Reporting Requirements.

Please submit within sixty days, proposed revisions to the Administrative Controls Section of your Technical Specifications which meet the guidance set forth in the enclosures to the extent practical for your facility. The proposed revision should be submitted in three signed and thirty-seven additional copies.

Although all of the provisions of Safety Guides 16 and 21 may not be included in the present Technical Specifications for your facility, we request that you immediately address the reporting format of the Guides, and, to the degree practical for your facility, make measurements and analyses equivalent to those specified in the Guides. Where you collect alternate, different, or additional data that should be reported separately, please report these data in units and style as similar as feasible to those of the Guides.

For those subjects addressed in the Safety Guides which you do not propose to adopt in your reports, we request that you submit the following additional information:

1. A description of those areas where it is not practical in your facility for you to follow measurements prescribed and the reporting format of the Guides.

2. A description of those measurements which depart either in frequency or in type of analysis from the acceptable program described in the Safety Guides and a justification for the adequacy of your alternate program.

Thank you for your cooperation in this effort to provide a consistent means of reporting information on operating data and effluent releases from nuclear power facilities.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Guide No. 16
2. Safety Guide No. 21
See Central File Copy
3. Plant Reporting Requirements

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

6.12 PLANT REPORTING REQUIREMENTS

6.12.1 The following information shall be submitted in addition to the reports listed in Table 6.12-1 and required by Title 10, Code of Federal Regulations.

6.12.2 Routine Reports:

a. Operating Reports

Operations Reports shall be submitted in writing to the Deputy Director for Reactor Projects, Directorate of Licensing, USAEC, Washington, D. C. 20545.

(1) Startup Report

A summary report of unit startup and power escalation testing shall be submitted following receipt of operating licenses, following amendments to the licenses involving a planned increase in power level, following the installation of fuel that has a different design and/or has been manufactured by a different fuel supplier, or following modifications to an extent that the nuclear, thermal, or hydraulic performance of the unit may be significantly altered. The report shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation should be described. Startup reports shall be submitted within 60 days following commencement of commercial power operation, i.e., initially following synchronization of the turbo-generator to produce commercial power or resumption of commercial power operation

(2) First Year Operation Report

A report shall be submitted within 60 days after completion of the first year of commercial power operation as defined above. This report may be incorporated into the semiannual operating report and shall cover the following:

(a) an evaluation of unit performance to date in comparison with design predictions and specifications;

(b) a reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analyses;

(c) an assessment of the performance of structures, systems and components important to safety;

(d) a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the plant, including items discussed in the AEC's safety evaluation, items on which additional information was required as

conditions of the license and items identified in the licensee's startup report.

(3) Semiannual Operating Reports

Routine operating reports shall be submitted within 60 days after January 1 and July 1 of each year. The first such period should begin with the date of initial criticality. These reports should include the following:

(a) Operations Summary

A summary of operating experience occurring during the reporting period that relates to the safe operation of the plant, including a summary of:

(i) changes in plant design,

(ii) performance characteristics (e.g., equipment and fuel performance),

(iii) changes in procedures which were necessitiated by (i) and (ii) or which otherwise were required to improve the safety of facility operations,

(iv) results of surveillance tests and inspections required by these technical specifications,

(v) the results of any periodic containment leak rate tests performed during the reporting period,

(vi) a brief summary of those changes, tests and experiments requiring authorization from the Commission pursuant to 10 CFR 50.59(a), and

(vii) any changes in the plant operating organization which involve positions which are designated as key supervisory personnel on Figure 6.2-1.

(b) Power Generation

A summary of power generated during the reporting period including:

(i) gross thermal power generated (in MWH)

(ii) gross electrical power generated (in MWH)

(iii) net electrical power generated (in MWH)

(iv) number of hours the reactor was critical

(v) number of hours the generator was on-line

(vi) histogram of thermal power vs. time

(c) Shutdowns

Descriptive material covering all outages occurring during the reporting period. For each outage, information shall be provided on:

(i) the cause of the outage,

(ii) the method of shutting down the reactor; e.g., trip automatic rundown, or manually controlled deliberate shutdown,

(iii) duration of the outage,

(iv) unit status during the outage; e.g., cold shutdown or hot shutdown,

(v) corrective action taken to prevent repetition, if appropriate.

(d) Maintenance

A discussion of safety-related maintenance (excluding preventative maintenance) performed during the reporting period on systems and components that are designated to prevent or mitigate the consequences of postulated accidents or to prevent the release of significant amounts of radioactive material. Included in this category are systems and components which are part of the reactor coolant pressure boundary defined in 10 CFR §50.2(v), any part of the engineered safety features, or associated service and control systems that are required for the normal operation of engineered safety features, part of any reactor protection or shutdown system, or part of any radioactive waste treatment handling and disposal system or other system which may contain significant amounts of radioactive material. For any malfunctions for which corrective maintenance was required, information shall be provided on:

(i) the system or component involved,

(ii) the cause of the malfunction,

(iii) the results and effect on safe operation,

(iv) corrective action taken to prevent repetition,

(v) precautions taken to provide for reactor safety during repair.

(e) Changes, Tests and Experiments

A summary of all changes in the plant design and procedures that relate to the safe operation of the plant shall be included in the Operations Summary section of these semiannual reports. Changes, tests, and experiments performed during the reporting period that require authorization from the Commission pursuant to 10 CFR 50.59(a) are covered in paragraph 6.12.1.a(3)(a)(vi) of these technical specifications; however, those changes, tests, and experiments that do not require Commission authorization pursuant to §50.59(a) shall be addressed. The report shall include a brief description and the summary of the safety evaluation for those changes, tests, and experiments, carried out without prior Commission approval, pursuant to the requirements of §50.59(b) of the Commission's regulations, that "The licensee shall furnish to the Commission,

annually or at such shorter intervals as may be specified in the license, a report containing a brief description of such changes, tests, and experiments, including a summary of the safety evaluation of each".

(f) Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from the plant, with data summarized on a monthly basis following the format of Appendix A of USAEC Safety Guide 21 of January 1972:

(i) Gaseous Effluents

(a) Gross Radioactivity Releases

(1) Total gross radioactivity (in curies), including noble and activation gases released.

(2) Maximum gross radioactivity release rate during any one-hour period.

(3) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(4) Percent of technical specification limit.

(b) Iodine Releases

(1) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(2) Percent of technical specification limit for I-131 released.

(c) Particulate Releases

(1) Gross radioactivity (β, γ) released (in curies) excluding background radioactivity.

(2) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(3) Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.

(4) Percent of technical specification limit for particulate radioactivity with half-lives greater than eight days.

(ii) Liquid Effluents

(a) Gross radioactivity (β, γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

(b) Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(c) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(d) Total volume (in liters) of liquid waste released.

(e) Total volume (in liters) of dilution water used prior to release from the restricted area.

(f) The maximum concentration of gross radioactivity (β, γ) released to the unrestricted area (averaged over the period of release).

(g) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

(h) Percent of technical specification limit and 10 CFR Part 20 concentration limits for unrestricted areas.

(iii) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet).

(b) The total estimated radioactivity (in curies) involved.

(c) The dates of shipment and disposition (if shipped offsite).

(g) Environmental Monitoring

(i) For each medium sampled, e.g., air, baybottom, surface water, soil, fish including:

(a) Number of sampling locations,

(b) Total number of samples,

(c) Number of locations at which levels are found to be significantly above local backgrounds,

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(ii) If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(iii) If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release

shall be provided.

(h) Occupational Personnel Radiation Exposure

A tabulation of personnel exposures shall be reported for the year (or first six months) in the following groups: less than 100 mRem, 100 - 500 mRem, 500 - 1250 mRem, 1250 - 2500 mRem, 2500 - 5000 mRem, above 5000 mRem. An explanation for all personnel exposures greater than 500 mRem in six months or the year shall be provided.

6.12.3 Non-Routine Reports

a. Reporting of Abnormal Events

(1) Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone or telegraph to the Director of the Regional Regulatory Operations Office, followed by a written report within 10 days to the Deputy Director for Reactor Projects, Directorate of Licensing (cc. to the Director of the Regional Regulatory Operations Office) in the event of the abnormal occurrences as defined in Section 1.0. The written report on these abnormal occurrences,

and to the extent possible, the preliminary telephone and telegraph notification, shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, and (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems.

In addition, the written report shall relate any failures or degraded performance of systems and components for the incident to similar equipment failures that may have previously occurred at the plant. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

b. Reporting of Unusual Events

A written report shall be forwarded within 30 days to the Deputy Director for Reactor Projects, Directorate of Licensing, and to the Director of the Regional Regulatory Operations Office, in the event of:

(1) Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the technical specifications.

(2) Any substantial variance from performance specifications contained in the technical specifications or in the Safety Analysis Report.

(3) Any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

6.12.4 Special Reports

Special reports shall be submitted in writing within 90 days to the Deputy Director for Reactor Projects, Directorate of Licensing, USAEC, Washington, D. C. 20545.

Special reports shall be submitted covering inspections, tests and maintenance that are appropriate to assure safe operation of the plant. The frequency and content of these special reports are determined on an individual case basis and designated in these technical specifications. Examples of subjects for such reports include:

(a) In-service inspection.

(b) Tendon surveillance.

(c) Containment structural tests.

(d) Special maintenance reports.

(e) Authorization of changes, tests, and experiments in accordance with 10 CFR 50.59.

(f) Containment leak rate tests.

(g) Radioactive effluent releases.

(h) Materials radiation surveillance specimens reports.

(i) Fuel performance following each refueling or partial refueling.

TABLE 6.12-1

REPORTING SUMMARY

Technical Specifications Paragraph	AEC Regulation	Report	Notification Within	Written Report Within[1]			
				10 Days	30 Days	6 Mo.	1 Yr.
6.6	20.403(a)	Severe Accident Involving Licensed Material	Immediately		DRO		
6.6	20.402	Loss of Licensed Material	Immediately		DRO		
6.6	73.42	Special Nuclear Material Unaccounted for	Immediately[2]				
6.6	40.64(c)	Theft or Unlawful Diversion of Source Material	Immediately[3]				
6.6	70.52	Accidental Criticality or Loss of Special Nuclear Material	Promptly				
6.6	70.54	Transfer of Special Nuclear Material	Promptly[4]				
		Receipt of Special Nuclear Material		AEC[4]			
6.6	40.64(a)	Transfer of Source Material	Promptly[4]				
		Receipt of Source Material		AEC[4]			
6.6	20.403(b)	Accidents Involving Licensed Material	24 hours		DRO		
6.6.2.a	50.36	Abnormal Occurrence	24 hours	DL/DRO			
6.6	20.405(a)	Overexposure or Excessive Radiation Level			DRO		
6.6.2.b	50.36	Unusual Events			DL/DRO		

TABLE 6.12-1 (Continued)

REPORTING SUMMARY

Technical Specifications Paragraph	AEC Regulation	Report	Notification Within	Written Report Within[1] 10 Days	30 Days	6 Mo.	1Y
6.6	20.408	Personnel Exposure (Terminated Employees)			DR[5]		
6.6	70.53	Special Nuclear Material Status				AEC[4]	
6.6.1.a(1)	50.36	Startup Report	DL[6]				
6.6.1.a(3)	50.36	Semiannual Operating				DL	
6.6.1.a(2)	50.36	First Year Operation					DL
6.6	20.407	Personnel Exposure and Monitoring					DR
6.6.3.e	50.59(d)	Changes, Test, and Experiments	DL[7]				
6.6.3.f	Proposed Appendix Part 50	Containment Leak Rate	DL[8]				

1 DR - Director of Regulation, DL - Directorate of Licensing, DRO - Directorate of Regulatory Operations

2 See 10CFR73.42 for details on reporting times.
3 See 10CFR40.64(c) for details on reporting times.
4 U.S. Atomic Energy Commission, P.O. Box E, Oak Ridge, Tennessee 37830.
5 Within 30 days after determining exposure or 90 days after termination, whichever is earlier.
6 Within 60 days following completion of testing or commencement of commercial power operation, whichever comes first.
7 AEC authorization is required prior to performing the change, test, or experiment in this category.
8 Report on first text due 3 months after completion of test; notification of other tests depends on test results.

DISTRIBUTION
W. Dooly, DR
RO (3)
H. Shapar, OGC
~~N. Dube, L (5)~~
J. R. Buchanan, ORNL
PDR
Local PDR
~~Docket File~~
RP Reading
Branch Reading
ACRS (16)
R. Boyd
R. DeYoung
D. Skovholt
T. Carter
R. Schemel
R. Woodruff
S. Teets

SEP 20 1972

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. N. B. Bessac, Manager
Nuclear Generation
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

Your report of an abnormal occurrence dated July 24, 1972, contains a description of action taken as a result of finding that the concentration of boron in the boron injection tank was below the limit specified in the Technical Specifications. We have reviewed your report and find that it does not contain sufficient information for us to evaluate the safety considerations involved. Please provide additional information as described in the enclosure within 10 days of receipt of this letter.

Sincerely,

/s/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

OFFICE ▶	L:OR-1	L	L:OR-1	L:OR		
SURNAME ▶	RWWoodruff:pl	SATeets	RJSchemel	DJSkovholt		
DATE ▶	9/19/72	9/19/72	9/19/72	9/ /72		

REQUEST FOR ADDITIONAL INFORMATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

Expand your discussion of the plugging of the boric acid recirculation line to include the following:

1. A description of the mechanism which caused the plugging.

2. A description of the means by which the low boron concentration was detected.

3. An estimate of the length of time the boron concentration could have been below the specified minimum.

4. A statement of the time required to correct the abnormal condition, the reactor operating status during this time, and justification for such status.

5. A description of action taken to prevent or reduce the probability of a recurrence.

6. A summary of the report prepared by the Plant Safety Committee subsequent to their review of the occurrence.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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RO (3)
H. Shapar, OGC
N. Dube, L (5)
J. Buchanan, ORNL
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Docket File
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Branch Reading
ACRS (16)
R. Boyd
R. DeYoung
J. Hendrie
E. Case
D. Knuth
D. Skovholt
T. Carter
R. Schemel
R. Woodruff, Craig Roberts
S. Teets, M. Jinks

SEP 12 1972

Docket No. 50-261

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

Gentlemen:

As a result of recent investigations of the performance of unprepressurized fuel in some of the pressurized water reactors presently in operation, we have determined that certain additional conditions are warranted for continued operation of the H. B. Robinson Unit No. 2. The basis for these conditions was discussed with your representatives on August 3, 1972.

Based on our review of the operating experience of H. B. Robinson Unit No. 2 and our discussions with you, we have concluded that operation of Unit No. 2 may be safely continued subject to the additional conditions specified in the enclosure to this letter entitled "Interim Conditions for Operation, H. B. Robinson Unit No. 2". These conditions shall be applicable to operation of Unit No. 2 at any authorized power level until the first refueling shutdown. Authorization of reactor operation under conditions other than those specified is dependent upon further evaluation of the longer term effects of the fuel anomalies associated with unprepressurized fuel.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Interim Conditions for Operation

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

O-9/14/72

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SURNAME ▶	RWWoodruff:pl SATeets	RJSchemel	DKnuth	ECase	JHendrie	DJSkovholt
DATE ▶	9/8/72	9/ 8 /72	9/ 11 /72	9/ [illegible] /72	9/ 11 /72	9/12/72

Form AEC-318 (Rev. 9-53) AECM 0240 GPO c43—16—81465-1 445-678

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 12, 1972

Docket No. 50-261

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

Gentlemen:

As a result of recent investigations of the performance of unprepressurized fuel in some of the pressurized water reactors presently in operation, we have determined that certain additional conditions are warranted for continued operation of the H. B. Robinson Unit No. 2. The basis for these conditions was discussed with your representatives on August 3, 1972.

Based on our review of the operating experience of H. B. Robinson Unit No. 2 and our discussions with you, we have concluded that operation of Unit No. 2 may be safely continued subject to the additional conditions specified in the enclosure to this letter entitled "Interim Conditions for Operation, H. B. Robinson Unit No. 2". These conditions shall be applicable to operation of Unit No. 2 at any authorized power level until the first refueling shutdown. Authorization of reactor operation under conditions other than those specified is dependent upon further evaluation of the longer term effects of the fuel anomalies associated with unprepressurized fuel.

Sincerely,

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosure:
Interim Conditions for Operation

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

INTERIM CONDITIONS FOR OPERATION

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

1. Average linear power in each unprepressurized fuel assembly shall not exceed 7.15 kW/ft.

2. For unprepressurized fuel assemblies, the peak linear power as obtained from evaluations of in-core flux maps and multiplied by 1.2 to allow for possible local power peaking, shall not exceed 13.2 kW/ft. For prepressurized fuel assemblies, the peak linear power obtained in the same way, shall not exceed 15.2 kW/ft.

3. Reactor shutdown will be initiated within 8 hr if the primary to secondary leakage in a steam generator reaches 0.3 gpm.

4. For cycle 1 operation:

   (a) In-core flux traces shall be measured and evaluated every 2 weeks. A full in-core flux map shall be measured and evaluated once a month.

   (b) Primary coolant gross radioactivity shall be measured at least five times per week and after each significant operating event which could affect fuel clad integrity.

   Primary coolant gross gamma radioactivity shall be monitored continuously by the letdown monitor. If the letdown monitor is not operating, the primary coolant gross activity shall be measured daily.

   (c) Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured at least daily to evaluate steam generator leak tightness.

5. A monthly report of all primary and secondary activity measurements and effluent discharge activity levels shall be made to the Directorate of Licensing.

6. For cycle 1 operation, the following administrative controls shall be applied:

   (a) Should a power level less than 95% be maintained continuously for more than 100 hours but less than 24 days, the rate of power increase shall be limited to 10% per hour.

   (b) Should a power level less than 95% be maintained continuously for more than 24 days, the rate of power increase shall be limited to 3% per hour from 25% to 100% of full power.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SEP 12 1972

Files (H. B. Robinson-2, Docket No. 50-261)
THRU: R. J. Schemel, Chief, ORB #1, L

SAFETY EVALUATION RE FUEL CLAD COLLAPSE

Axial flux traverses indicate that separation of fuel pellets and collapse of the clad into the resulting void may be in progress in Region 1 which contains unprepressurized fuel. Analyses have been performed by Westinghouse for Carolina Power & Light Company which indicate that (a) limiting linear power density in the unprepressurized fuel to 13.2 kW/ft, and in the prepressurized fuel to 15.2 kW/ft will limit temperature of the cladding to $1800^{o}F$ and $2300^{o}F$ respectively if a loss-of-coolant accident were to occur, (b) limiting the leakage from the steam generators to 0.3 gpm each would limit the thyroid dose at the site boundary to values on the order of Part 20 limits if the leakage is confined to a single steam generator, and (c) limiting the rate of power increase will limit the consequences of differential thermal expansion of cladding and fuel.

CP&L has not formally submitted descriptions of the analyses and the results and conclusions obtained from them; however, after many discussions between the staff and CP&L and their consultants, Technical Review has informed us that they believe that the Interim Conditions for Operation are appropriate for continued operation while the demand for power is great. Technical Review is completing its analytical model of the fuel and will apply the model to Robinson-2 fuel.

R. W. Woodruff

R. W. Woodruff
Operating Reactors Branch #1
Directorate of Licensing

cc: A. Giambusso
D. Knuth
D. J. Skovholt OK DJK 9/12/72
T. J. Carter
R. J. Schemel
R. W. Woodruff (2)
S. Teets
RO (3)
M. Jinks (2)

JUL 28 1972

Docket No. 50-261

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

Gentlemen:

Requirements for peaking factor reductions as a result of recent evaluations of emergency core cooling system performance and a recent evaluation of the ability of the ex-core detectors to detect local power perturbations caused by a misaligned rod or a dropped rod indicates that present quadrant power tilt limits in the Technical Specifications for the H. B. Robinson plant may be too large, permitting design peaking factors or safety limits to be exceeded before detection by the ex-core instrumentation. These limits on the quadrant power tilt are specified in the Control Rod and Power Distribution Limits section of your Technical Specifications. Two tilt limits, as determined by ex-core detectors, are specified: (1) a lower limit that provides a warning of potential violation of design peaking factors, and (2) an upper limit that provides a warning of potential violation of safety limits. Action appropriate to each situation is also specified.

In this regard, please submit by August 25, 1972, a reevaluation of the ability of the ex-core detectors to provide a warning of potential violation of design peaking factors and safety limits from the x-y plane power tilts for the H. B. Robinson plant assuming the most adverse permissible axial peaking factor. If the results of your analysis show that lower quadrant tilt limits are needed to provide the above warnings, we will require appropriate changes to the Technical Specifications. In this event, your response should include your proposed changes to the

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Technical Specifications. Please inform us within seven (7) days after receipt of this letter of your confirmation of the above submittal date or the date you will be able to meet.

If you desire any discussion or clarification of the material requested, please contact us.

Sincerely,

D. J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

cc: G. F. Trowbridge
Shaw, Pittman, Potts, Trowbridge & Madden
910 - 17th Street, N.W.
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ACRS(16)

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DATE ▶	7/26/72	7/27/72	7/28/72	7/28/72		

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RO (3)
R. Woodruff
Lic. Asst. (2)
(Stella Teets)

Docket No. 50-261

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

Gentlemen:

Requirements for peaking factor reductions as a result of recent evaluations of emergency core cooling system performance indicate that present quadrant power tilt limits in the Technical Specifications for the H. B. Robinson plant may be too large, permitting design peaking factors or safety limits to be exceeded before detection by the ex-core instrumentation. These limits on the quadrant power tilt are specified in the Control Rod and Power Distribution Limits section of your Technical Specifications. Two tilt limits, as determined by ex-core detectors, are specified: 1) a lower limit that provides a warning of potential violation of design peaking factors, and 2) an upper limit that provides a warning of potential violation of safety limits. Action appropriate to each situation is also specified.

In this regard, please submit by August 25, 1972, a reevaluation of the ability of the ex-core detectors to provide a warning of potential violation of design peaking factors and safety limits from the x-y plane power tilts for the H. B. Robinson plant assuming the most adverse permissible axial peaking factor. If the results of your analysis show that lower quadrant tilt limits are needed to provide the above warnings, we will require appropriate changes to the Technical Specifications. In this event, your response should include your proposed changes to the Technical Specifications. Please inform us within seven (7) days after receipt of this letter of your confirmation of the above submittal date or the date you will be able to meet.

If you desire any discussion or clarification of the material requested, please contact us.

Sincerely,

D. J. Skovholt, Assistant Director
for Operating Reactors
~~Directorate of Licensing~~

cc: See next page

OFFICE ▶	cc: See next page	OR:L	OR:L	TR:L	OR:AD:L	
SURNAME ▶	CHall	RWoodruff:	RSchemel	VMoore	DSkovholt	
DATE ▶	7/	7/ /72	7/ /72	7/ /72	7/ /72	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1971 -446 -154

cc: G. F. Trowbridge
Shaw, Pittman, Potts,
Trowbridge & Madden
910 - 17th Street, N.W.
Washington, D. C. 20006

OFFICE ▶						
SURNAME ▶						
DATE ▶						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-261

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

and a recent evaluation of the ability of the ex-core detectors to detect local power perturbations caused by a misaligned rod or a dropped rod R.W.L.

Gentlemen:

Requirements for peaking factor reductions as a result of recent evaluations of emergency core cooling system performance indicate that present quadrant power tilt limits in the Technical Specifications for the H. B. Robinson plant may be too large, permitting design peaking factors or safety limits to be exceeded before detection by the ex-core instrumentation. These limits on the quadrant power tilt are specified in the Control Rod and Power Distribution Limits section of your Technical Specifications. Two tilt limits, as determined by ex-core detectors, are specified: 1) a lower limit that provides a warning of potential violation of design peaking factors, and 2) an upper limit that provides a warning of potential violation of safety limits. Action appropriate to each situation is also specified.

In this regard, please submit by August 25, 1972, a reevaluation of the ability of the ex-core detectors to provide a warning of potential violation of design peaking factors and safety limits from the x-y plane power tilts for the H. B. Robinson plant assuming the most adverse permissible axial peaking factor. If the results of your analysis show that lower quadrant tilt limits are needed to provide the above warnings, we will require appropriate changes to the Technical Specifications. In this event, your response should include your proposed changes to the Technical Specifications. Please inform us within seven (7) days after receipt of this letter of your confirmation of the above submittal date or the date you will be able to meet.

If you desire any discussion or clarification of the material requested, please contact us.

Sincerely,

D. J. Skovholt, Assistant Director
for Operating Reactors
Directorate of Licensing

cc: See next page

APR 6 - 1972

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Docket File
DRL Reading
Branch Reading
ACRS (16)
D. Skovholt
T. Carter
J. McGough
R. Schemel
R. Woodruff
T. Wambach
S. Teets

Docket No. 50-261

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

Gentlemen:

On March 3, 1972, you submitted WCAP 7844 "Plant Startup Test Report" for the H. B. Robinson facility. In this report, you discuss modifications made to the residual heat removal system. It is reported that these modifications were required as the result of pump runout problems and required the addition of a bypass line around the RHR discharge line leading to the reactor coolant system. In addition, other changes involving system operation were determined to be required based on later analyses.

We are concerned that these modifications may significantly affect the stated design capability, operational reliability, and redundancy characteristics required of this system for safe facility operation. Additionally, we believe these modifications made without our authorization may constitute an unreviewed safety question as defined by 10 CFR 50.59(c) or may have warranted revisions to the technical specifications, such as revised or additional surveillance requirements.

In order for us to resolve this issue in a prompt and thorough manner, please provide us the information requested in the enclosure by April 21, 1972. Information submitted should be of sufficient detail and clarity to allow our staff to evaluate the adequacy of the system modifications.

If you desire further information, please contact Mr. Robert J. Schemel at 301-973-7433.

Sincerely,

/s/

Donald J. Skovholt
Assistant Director
for Reactor Operations
Division of Reactor Licensing

Enclosure:
Request for Information on Residual Heat Removal System Modifications

D-4/7/72

OFFICE ▶	DRL	DRL	DRL	DRL	DRL	
SURNAME ▶	JMMcGough:pdl	TWambach	SATeets	RJSchemel	DJSkovholt	
DATE ▶	4/5/72	4/6/72	4/5/72	4/6/72	4/6/72	

REQUEST FOR INFORMATION ON RESIDUAL

HEAT REMOVAL SYSTEM MODIFICATIONS

H. B. ROBINSON

DOCKET NO. 50-261

1. Provide the performance specifications for the RHR system. Identify the minimum flow rate and pump head required for normal decay heat removal and for core reflooding during LOCA conditions.

2. Provide a chronological summary of the modifications made to the RHR system since submittal of the FSAR. Include the dates that the modifications and the system flow tests were performed. Include the dates that the Safety Committee approval was given to each modification.

3. With regard to the original modification involving the installation of the orificed bypass line around HCV-758, provide the following:

   3.1 the bases for initiating the modifications including data, test results, or predictions indicating the severity of the pump runout problem.

   3.2 actual and/or predicted pump curves and system flow rates for single and dual pump operation before and after completion of initial modification.

4. With regard to the revised modification involving blanking off the orificed bypass line around HCV-758, provide the following information:

   4.1 the bases and safety evaluation associated with this modification including provisions taken to insure reliability of HCV-758 to maintain its throttled position.

   4.2 results of system testing performed to demonstrate system flow characteristics. Include system flow rates and pump head data for single and dual RHR pump operation. Indicate where pump flow data falls on pump performance curve. Provide flow rate data versus HCV-758 position for single pump operation.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

5. Describe the anticipated final revision you intend to make to this system and indicate your schedule for completion. Discuss whether this revision requires Commission approval, under 10 CFR 50, either as an unreviewed safety question or by warranting a change to the technical specifications. Provide a summary of the system testing you will perform upon completion of modifications including provisions for long term recirculation tests to insure pump motor reliability.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

APR 6 - 1972

Docket No. 50-261

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

Gentlemen:

On March 3, 1972, you submitted WCAP 7844 "Plant Startup Test Report" for the H. B. Robinson facility. In this report, you discuss modifications made to the residual heat removal system. It is reported that these modifications were required as the result of pump runout problems and required the addition of a bypass line around the RHR discharge line leading to the reactor coolant system. In addition, other changes involving system operation were determined to be required based on later analyses.

We are concerned that these modifications may significantly affect the stated design capability, operational reliability, and redundancy characteristics required of this system for safe facility operation. Additionally, we believe these modifications made without our authorization may constitute an unreviewed safety question as defined by 10 CFR 50.59(c) or may have warranted revisions to the technical specifications, such as revised or additional surveillance requirements.

In order for us to resolve this issue in a prompt and thorough manner, please provide us the information requested in the enclosure by April 21, 1972. Information submitted should be of sufficient detail and clarity to allow our staff to evaluate the adequacy of the system modifications.

If you desire further information, please contact Mr. Robert J. Schemel at 301-973-7433.

Sincerely,

Donald J. Skovholt

Donald J. Skovholt
Assistant Director
for Reactor Operations
Division of Reactor Licensing

Enclosure:
Request for Information on Residual
Heat Removal System Modifications

REQUEST FOR INFORMATION ON RESIDUAL

HEAT REMOVAL SYSTEM MODIFICATIONS

H. B. ROBINSON

DOCKET NO. 50-261

1. Provide the performance specifications for the RHR system. Identify the minimum flow rate and pump head required for normal decay heat removal and for core reflooding during LOCA conditions.

2. Provide a chronological summary of the modifications made to the RHR system since submittal of the FSAR. Include the dates that the modifications and the system flow tests were performed. Include the dates that the Safety Committee approval was given to each modification.

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   3.1 the bases for initiating the modifications including data, test results, or predictions indicating the severity of the pump runout problem.

   3.2 actual and/or predicted pump curves and system flow rates for single and dual pump operation before and after completion of initial modification.

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   4.1 the bases and safety evaluation associated with this modification including provisions taken to insure reliability of HCV-758 to maintain its throttled position.

   4.2 results of system testing performed to demonstrate system flow characteristics. Include system flow rates and pump head data for single and dual RHR pump operation. Indicate where pump flow data falls on pump performance curve. Provide flow rate data versus HCV-758 position for single pump operation.

5. Describe the anticipated final revision you intend to make to this system and indicate your schedule for completion. Discuss whether this revision requires Commission approval, under 10 CFR 50, either as an unreviewed safety question or by warranting a change to the technical specifications. Provide a summary of the system testing you will perform upon completion of modifications including provisions for long term recirculation tests to insure pump motor reliability.

Docket No. 50-261

FEB 4 1972

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

Gentlemen:

We have evaluated your report "H. B. Robinson Unit No. 2 Emergency Core Cooling Performance" dated September 29, 1971, and supplements thereto, dated November 5 and December 8, 1971.

On the basis of our review of the information you have provided and of comparative studies performed on other plants of similar configuration, we have concluded that your proposal to deactivate the automatic hot leg injection capability will not present significant safety hazards not described or implicit in the safety analysis report. We request that you provide the details of how you intend to implement this modification.

The information presented in your most recent submittal in support of your small break analyses appears to be inconsistent with previous analyses that you have furnished. Some specific items noted include (1) the volumes corresponding to fluid levels at the top and bottom of the reactor core as indicated in the December 8, 1971, submittal differ from those indicated in your January 25, 1971, and August 12, 1970, submittals and (2) a comparison of core volume transients with and without hot leg injection for 6, 4 and 3-inch diameter breaks indicates a higher core water level with two less lines delivering flow to the reactor.

We request that you evaluate these apparent inconsistencies and provide an explanation for them, so that the record will reflect a complete and correct evaluation of the performance of the emergency core cooling system.

LB

OFFICE ▶						
SURNAME ▶						
DATE ▶						

FEB 4 1972

The requested information should be provided as soon as possible. Within seven days please inform us of your schedule for submission of this information.

Please contact us if you desire to discuss this matter further.

Sincerely,
Original Signed by
P. A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

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VHWilson, DRL (2)
JMMcGough, DRL
ACRS (16)

OFFICE ▶	DRL:PWR-3	DRL:AD/PWRs	DRL:D/DIR	DRL:DIR		
RNAME ▶	x7415 KRG KRGoller	RCDeYoung	FSchroeder	PAMorris		
▶	2/2/72	2/ /72	2/2/72	2/3/72		

Docket No. 50-261

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

Gentlemen:

We have evaluated your report "H. B. Robinson Unit No. 2 Emergency Core Cooling Performance" dated September 29, 1971, and supplements thereto, dated November 5 and December 8, 1971.

We have concluded on the basis of our review of the information you have provided and of ~~in conjunction with~~ comparative studies performed on other plants of similar configuration that your proposal to deactivate the automatic hot leg injection capability ~~is acceptable. Such a modification~~ will not ~~in our opinion~~ present significant safety hazards not described or implicit in the safety analysis report. ~~Accordingly,~~ We request that you provide the details of how you intend to implement this modification.

~~Additionally, we wish to call your attention to the fact that~~ the information presented in your most recent submittal in support of your small break analyses appears to be inconsistent with previous analyses that you have furnished. Some specific items noted include (1) the volumes corresponding to fluid levels at the top and bottom of the reactor core as indicated in the December 8, 1971, submittal differ from those indicated in your January 25, 1971, and August 12, 1970, submittals and (2) a comparison of core volume transients with and without hot leg injection for 6, 4 and 3-inch diameter breaks indicates a higher core water level with two less lines delivering flow to the reactor.

We request that you evaluate these apparent inconsistencies and provide an explanation for them, ~~These inconsistencies and the overall lack of definitive supporting information accompanying your request should be corrected at the earliest opportunity in order to assure that the documented bases for operation of your facility are complete and correct, and to demonstrate management's awareness that a complete and correct record is a necessary requisite~~

so that the record will reflect a complete and correct evaluation of the performance of the emergency core cooling system.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

~~for safe operation. In our opinion the necessary corrections should be able to be made in a relatively short time. Accordingly, we will require~~ this matter should ~~to~~ be resolved to our satisfaction no later than February 15, 1972. In the event that it is not so resolved and unless we ~~cannot~~ extend the time for resolution for good cause, we will ~~intend to~~ consider the need for restrictions on operation to provide the needed assurance that the facility is being operated in a safe manner.

Please contact us if you desire ~~We are prepared~~ to discuss this matter further ~~with you at your convenience.~~

Sincerely,

P. A. Morris, Director
Division of Reactor Licensing

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

The requested information should be provided as soon as possible. Please inform us within seven days of your schedule for submission of this information.

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EGCase, DRS
RRMaccary, DRS
PWR Branch Chiefs
OGC
CO (2)
VHWilson, DRL (2)
JMMcGough, DRL
ACRS (16)

OFFICE ▶	DRL:PWR-3	DRL:AD/PWRs	DRL:D/DIR	DRL:DIR		
SURNAME ▶	x7415 KRG KRGoller:esp	RCDeYoung	FSchroeder	PAMorris		
DATE ▶	1/31/72	1/ /72	1/ /72	1/ /72		

PWR Branch Chiefs
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CO (2)
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ACRS (16)

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RWKlecker, DRL
EGCase, DRS
RRMaccary, DRS

Docket No. 50-261

JAN 31 1972

License No. DPR-23

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

Gentlemen:

As you are aware, an event occurred at a foreign pressurized water power reactor in which an unusual corrosion mechanism occurred when prolonged leakage of borated reactor coolant onto the reactor vessel head was undetected. Subsequent tests have indicated that this corrosion potential might exist under certain conditions when borated fluid has prolonged contact with carbon steel.

To preclude additional experiences of this type, an appropriate program of inservice inspection should be implemented to detect such effects at an early stage. The ASME Code Committee for Inservice Inspection is considering revision to the ASME Code for Inservice Inspection of Nuclear Reactors. However, as an interim measure, we believe that the inspection program described in the enclosure should be incorporated into your inservice inspection program.

Please advise us within thirty days concerning your adoption of the provisions of the enclosure.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
PWR Inservice Inspection
Program

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
~~910 17th Street, N. W.~~
Washington, D. C. 20006

OFFICE ▶			DRL:PWR-3	DRL:PWR-3	DRL:AD/PWRs	
SURNAME ▶			x7415 JMMcGough:ep	KRG KRGoller	RCDeYoung	
DATE ▶			1/28/72	1/28/72	1/28/72	

Form AEC-318 (Rev. 9-53)　U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

Recommended PWR Inservice Inspection Program for Detection of Effects of Reactor Coolant Leakage

A. Inspection Requirements

(1) Prior to reactor startup following each refueling outage, all pressure-retaining components of the reactor coolant pressure boundary shall be visually examined for evidence of reactor coolant leakage while the system is under a test pressure not less than the nominal system operating pressure at rated power.

This examination (which need not require removal of insulation) shall be performed by inspecting (a) the exposed surfaces and joints of insulations, and (b) the floor areas (or equipment) directly underneath these components.

At locations where reactor coolant leakage is normally expected and collected (e.g., valve stems, etc.), the examination shall verify that the leakage collection system is operative and leak-tight.

(2) During the conduct of the examinations of (1) above, particular attention shall be given to the insulated areas of components constructed of ferritic steels to detect evidence of boric acid residues resulting from reactor coolant leakage which might have accumulated during the service period preceding the refueling outage.

(3) The visual examinations of (1) and (2) above shall be conducted in conformance with the procedures of Article IS-211 of Section XI of the ASME Boiler and Pressure Vessel Code.

B. Corrective Measures

(1) The source of any reactor coolant leakage detected by the examinations of A(1) above shall be located by the removal of insulation where necessary and the following corrective measures applied:

(a) Normally expected leakage from component parts (e.g., valve stems) shall be minimized by appropriate repairs and maintenance procedures. Where such leakage may reach the surface of ferritic components of the reactor coolant pressure boundary, the leakage shall be suitably channeled for collection and disposal.

(b) Leakage from through-wall flaws in the pressure-retaining membrane of a component shall be eliminated, either by corrective repairs or by component replacement. Such repairs shall conform with the requirements of Article IS-400 of Section XI of the ASME Boiler and Pressure Vessel Code.

(2) In the event boric acid residues are detected by the examinations of A(2) above, insulation from ferritic steel components shall be removed to the extent necessary for examination of the component surfaces wetted by reactor coolant leakage to detect evidence of corrosion.

The following corrective measures shall be applied:

(a) An evaluation of the effect of any corroded area upon the structural integrity of the component shall be performed in accordance with the provisions of Article IS-311 of Section XI Code.

(b) Repairs of corroded areas, if necessary, shall be performed in accordance with the procedures of Article IS-400 of Section XI Code.

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PWR-3 Reading
HDenton
VHWilson (2)

OCT 14 1971

Docket No. 50-261

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

Gentlemen:

A copy of a letter, dated October 6, 1971, from the U. S. Department of the Interior, Fish and Wildlife Service, concerning the environmental monitoring program for H. B. Robinson, Unit No. 2, is enclosed for your information.

We particularly invite your attention to the following comment and recommendation by the Fish and Wildlife Service:

> "The monitoring program, however, does not agree with all of our previous recommendations. Our concern is that levels of radiation exposure from radioactive effluents may reach levels that will damage aquatic life. Therefore, it is important that radiological samples include benthic organisms, aquatic plants, and fish in addition to water and sediments.
>
> "We recommend that the above be included in the environmental monitoring program for this station."

Sincerely,

Original Signed by
P. A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Fish & Wildlife ltr dtd 10/6/71

cc w/encl:
See Attached

OFFICE ▶	AD:PWR	PWR-3	AD:PWR	DIR:DRL		
SURNAME ▶	VHW VHWilson/sb	JMcGough KGoller KRG	KRG/for RCDeYoung	M PAMorris		
DATE ▶	10/14/71	10/14/71	10/14/71	10/14/71		

OCT 14 1971

cc w/encl:

George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 Seventeenth Street, N. W.
Washington, D. C. 20006

cc w/o encl:

Mr. Daniel W. Slater, Chief
Division of River Basin Studies
Bureau of Sport Fisheries and Wildlife
U. S. Department of the Interior
Washington, D. C. 20240

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket

Docket No. 50-261 OCT 1 1971

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

Gentlemen:

This is in response to your letter of September 10, 1971, in which you pointed out that we had inadvertently referred to Carolina Power & Light Company's operating license as "Provisional Operating License No. DPR-23" instead of "Facility Operating License No. DPR-23" in our letters of May 27, and September 1, 1971. Please substitute the word "Facility" for "Provisional" in the above mentioned letters and also in our April 1, 1971, letter to you.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 Seventeenth Street, N. W.
Washington, D. C. 20006

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OGC (2)
Compliance
JRBuchanan
TWLaughlin, DTIE
RSBrodsky, NR
JLCaplin, RPS
JMMcGough, DRL

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SURNAME ▶	VHWilson/slb	JMMcGough	KRG KRGoller	RCDeYoung	PAMorris	
DATE ▶	9/27/71	9/27/71	9/29/71	9/29/71	9/29/71	

JUL 9 1971

Docket No. 50-261

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

Gentlemen:

On June 19, 1971, the AEC adopted interim acceptance criteria for the performance of emergency core cooling systems (ECCS) in light-water nuclear power plants. A copy of the Commission's interim policy statement on this matter is enclosed for your information. In accordance with Section IV.C.1.(a) of the interim policy statement you are requested to submit analyses of the performance of the ECCS presently installed in the H. B. Robinson Unit 2 facility using methods equivalent to the evaluation model in Appendix A, Part 3 of the interim policy statement as soon as practicable, but not later than October 1, 1971, to show that the performance of the ECCS is in compliance with the criteria of Sections IV.A. and B. of the statement. We have discussed this request with representatives of the Westinghouse Electric Corporation and we understand that appropriate analyses have been or are being performed for your plant.

The information that we need regarding the analyses is:

(1) For the break size range, location and type mentioned in Appendix A, Part 3, of the interim policy statement, provide information pertaining to (a) the system pressure, (b) the hot-spot clad temperature, local mass velocity, fluid temperature, and heat transfer coefficient, (c) the core pressure drop, quality, and mass velocity, (d) the heat flux distribution in the hot channel, (e) the flow rates in the upper and lower plenums, (f) the flow rates in the broken and intact cold-leg and hot-leg piping, (g) the flow rate out of the break, and (h) percent clad metal-water reaction.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

(2) Provide a detailed discussion of the calculation used to predict heat transfer during the reflood portion of the transient.

(3) Discuss in detail any deviations in the evaluation model used in the foregoing studies from that described in Appendix A, Part 3 of the interim policy statement.

In addition, you should submit for our review any changes to the Technical Specifications of License No. DPR-23 that may be required on the basis of the results of your analyses.

When this information has been prepared please send us 60 copies. When we have completed our review of this information, we plan to contact you regarding the results of our evaluation.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
AEC Interim Policy Statement

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge, & Madden
910 Seventeenth Street, N.W.
Washington, D.C. 20006

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SURNAME ▶	GOller/gl	RCDeYoung		PAMorris		
DATE ▶	6/30/71	6/30/71	/ /71	7/9/71		

Docket

Docket No. 50-261 MAR 29 1971

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

Gentlemen:

We have noted that the emergency diesel generators did not start or operate reliably at the H. B. Robinson facility on several occasions during the startup and power ascension test programs. We have also become aware that the protective trip devices associated with these machines are more extensive in number and type than those described in Section 8.2.3 of your Final Safety Analysis Report.

In view of our continuing interest in the reliable operation of this facility, please furnish us with the following information:

1. A complete description, including setpoint and expected margin-to-trip, of each electrical, mechanical or process type protective trip function which will prevent startup or initiate shutdown of the diesel generators.

2. The basis for requirement of each protective device, including a discussion of the factors involved in removal or replacement by alarm circuits.

3. An estimate of the time interval from each trip setpoint to diesel and/or generator failure assuming the trip is not activated and the unit continues operation.

4. A description of all alarm or trip indications associated with emergency generator operation including location and type of readout. Specify whether each unit and function is indicated separately or if a common alarm/trip system is used.

5. Specify if any of the protective devices are capable of being bypassed during either manual, test, or emergency (i.e., normal) operation. Describe the usage of such bypasses if applicable, including location and administrative controls.

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6. A detailed description with diagrams of the diesel cooling system and its operation including auxiliary circulating water pumps, heat exchangers, power sources and location of pressure control valves.

7. A detailed description with diagrams of the diesel lubricating oil system and its operation including auxiliary lube oil pumps, power sources and timing relay operation during diesel startup.

8. A detailed description with diagrams of the compressed air start system and its operation including receiver tanks, compressor valving, piping and drive detail and associated instrumentation.

If you desire further clarification or discussion of this matter, please contact us.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 Seventeenth Street, N. W.
Washington, D. C. 20006

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Docket No. 50-261

JAN 1 5 1971

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

Gentlemen:

This letter refers to your November 12, 1970, request for authorization to make allowance for the use of respiratory protective equipment in determining exposures for individuals exposed to radioactive material concentrations in excess of the limits specified in Appendix B, Table 1, Column 1, of 10 CFR 20.

As indicated to you in subsequent telephone conversations, your request should be resubmitted as a proposed change to the Technical Specifications for the H. B. Robinson Plant. As a basis for your request, you should include the details and methods of implementing the respiratory protection program.

We also note that the protection factors you requested are consistent with those published as a Proposed Rule in the Federal Register of November 4, 1967. The Proposed Rule has not been made effective. Comments we received and our additional evaluation of the Proposed Rule indicate that the protection factors in the published Proposed Rule may not be sufficiently conservative for use at this time. We suggest that you review your request in this regard and where possible, employ appropriate process and engineering controls to limit the concentrations of radioactive materials so that the necessary personnel protection would be provided by lower protection factors such as those we have previously approved for Big Rock Point (Docket No. 50-155) and Point Beach (Docket No. 50-266).

If you desire further clarification or discussion of this matter, please contact us.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

cc: See attached

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SURNAME ▶	JMMcGough:esp	KRGoller	RCDeYoung	FSchroeder	PAMorris	
DATE ▶	1/13/71	1/13/71	1/13/71	1/14/71	1/14/71	

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cc: George F. Trowbridge, Esq.
Shaw, Pittman, Potts, Trowbridge & Madden
910 17th Street, N. W.
Washington, D. C. 20006

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SURNAME ▶						
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