



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

January 10, 1977

FILE

Docket No. 50-261

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

Gentlemen:

RE: H. B. ROBINSON UNIT NO. 2

In August of this year we sent letters to you and the licensees of other operating PWR facilities which expressed our concern over the number of reported instances of reactor vessel overpressurization. We requested that an analysis be provided of the reactor coolant system (RCS) response to pressure transients and that any design modifications be identified that were determined to be necessary to preclude exceeding the limits of Appendix G to 10 CFR Part 50. In addition, we requested that if those design changes could not be made within a few months, interim measures should be implemented immediately to reduce the likelihood of overpressurization events until the permanent design changes can be made.

In your letter of October 27, 1976, you identified the interim measures that were being implemented at your facility. We have completed our review of your submittal and have concluded that additional information will be required for us to determine the adequacy of these interim measures. In addition, our review of the responses from other PWR licensees has identified certain measures which, in our opinion, are significant enough to be required in most PWR facilities. These measures have been identified as staff positions and are included with the additional information requests in the enclosure to this letter. In addition, we have concluded that the procedural and administrative measures you have already instituted will help prevent any future pressure transients and should be continued even after long term hardware improvements are made unless you can demonstrate that this would not be justified.

With regard to your submittal of December 16, 1976, in which a description and schedule information related to your "Reference Mitigating System" for long-term overpressure protection were provided, we find your proposed schedule to be unacceptable and your proposed course of action to be lacking with respect to meeting the design criteria detailed at our November 4, 1976 meeting (meeting summary attached). We called the November meeting to inform all PWR licensees of the specific design criteria that would be required. The response in your December letter is based on your interpretation of the preliminary criteria issued in August and does not adequately address the final criteria as detailed at the November meeting. The overpressure

8410310140 770110  
PDR ADCK 05000261  
P PDR

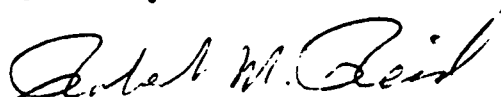
protection system you have proposed, based on your interpretation of the preliminary criteria may be sufficient as an interim remedy, but will not be acceptable for the long term until the final criteria are addressed and any deviations justified. In addition, the extended schedule you have proposed for analysis and acquisition of hardware will not meet the objective of implementation of improved overpressure protection in all operating PWR facilities by the end of 1977. Therefore, to meet this objective you should select either of the following two implementation options:

1. Commitment to a schedule for the installation of acceptable long-term improvements by December 31, 1977, which meet all the design criteria discussed at the November 4, 1976 meeting held in Bethesda, or
2. Commitment to a schedule to achieve installation of interim hardware improvements by December 31, 1977, and installation of long term improvements that meet all the design criteria discussed at the November meeting during the first scheduled shutdown after December 31, 1977. The dual setpoint system of pressurizer relief valves described in the Westinghouse "Reference Mitigating System" design is an acceptable interim hardware improvement.

The long term hardware improvements should either meet the design criteria discussed at the November 4, 1976 meeting; or, where deviations from the criteria are proposed, a detailed justification should be provided. Interim improvements need not meet all the design criteria discussed at the November meeting, but must represent good engineering practice and must not adversely affect plant safety or introduce potential common mode failures that could both cause the overpressure event and disable the protection system. For example, if air operated dual setpoint pressurizer relief valves are used as an interim improvement, air accumulators on the actuation mechanism and alarms to indicate loss of instrument air should be provided to insure that protection is provided and to alert the operator in the event that instrument air is lost. Whichever implementation option you select, your submittal detailing the proposed design improvements must be submitted in sufficient time to allow our review and approval to meet the objective discussed above.

You are requested to identify to us the implementation option you have selected and to provide your response to the staff positions and requests for additional information identified in the attached enclosure, within 45 days of receipt of this letter.

Sincerely



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

Enclosures and cc:  
See next page

Carolina Power & Light Company - 3 -

Enclosures:

1. Positions and Additional  
Information Requests
2. Meeting Summary

cc w/enclosures: See next page

Carolina Power & Light Company

cc: G. F. Trowbridge, Esq.  
Shaw, Pittman, Potts & Trowbridge  
1800 M Street, N. W.  
Washington, D. C. 20036

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

Chief, Energy Systems  
Analyses Branch (AW-459)  
Office of Radiation Programs  
U. S. Environmental Protection Agency  
Room 645, East Tower  
401 M Street, S.W.  
Washington, D.C. 20460

U. S. Environmental Protection Agency  
Region IV Office  
ATTN: EIS COORDINATOR  
345 Courtland Street, NE  
Atlanta, Georgia 30308

ENCLOSURE 1

STAFF POSITIONS AND ADDITIONAL INFORMATION REQUESTS

H. B. ROBINSON UNIT NO. 2

DOCKET NO. 50-261

1. The staff considers it essential that all plant operators (i.e., reactor operators, equipment operators, Instrument & Control personnel) be made aware of the details of the pressure transients which have taken place at all PWR facilities. POSITION: Formal discussions should be held with the operator to review the causes of past pressure transients that have occurred at other operating PWR facilities. Your discussions should include the plant conditions at the time, the mitigating action that could have been or was taken, and the preventive measures that could have been taken to avoid the event and the steps taken to prevent similar, further occurrences. Plant similarities and distinctions should be identified along with how these relate to plant startup, shutdown, and testing operations. With regard to this position, you are requested to provide the following information:
  - a. If you have not already completed the required formal discussion, when will you do so?
  - b. How will the discussions be held?
  - c. Of the past PWR Appendix G violations that have occurred at PWR facilities and which are described in License Event Reports, identify which are not credible in your plant due to equipment differences. Provide a description of the distinctions.
  - d. Describe, in detail, how you are reducing the likelihood of the other remaining credible events. Furnish schematics, diagrams or procedural summaries necessary to support the effectiveness and reliability of these measures.
  
2. The majority of the reported pressure transients events have occurred while the plants were operating in a water solid condition. POSITION: The staff will require that operations during which the plant is maintained in a water solid condition be minimized or if possible eliminated. Those operations in which the plant is in a water solid condition must be fully justified. Accordingly, please provide the following information:
  - a. Describe the procedures, evolutions or situations that require the plant be maintained in a water solid condition. Also provide reasons why a nitrogen, air or steam bubble cannot be maintained in these situations.

- b. Include sufficient background or supplementary information such as system diagrams, procedure summaries and descriptions of equipment operation to justify your need for operating the plant in a water solid condition.
3. The inadvertent operation of SIS components during cold shutdown conditions has been responsible for a major portion of the over-pressure incidents. POSITION: Based on the licensee submittals, the recent November 3-5, 1976 meetings, and discussions with NSSS vendors, the staff will require the deenergizing of SIS pumps and closure of SI header/discharge valves during cold shutdown operations. Those situations during which this position cannot be met must be described and be fully justified. Accordingly, please provide the following information:
- a. A schematic diagram of the SIS showing the flowpaths into the RCS.
  - b. The head-flow characteristics of each of the SIS pumps.
  - c. Identify on the schematic diagrams the pumps and the valves to be closed and deenergized.
  - d. Your time schedule for implementing these administrative and operating procedural changes to meet this position.
  - e. Indicate all circumstances for which the SIS pumps and valves may not be isolated and deenergized and for those situations, describe the manner in which SIS injection would be prevented.
  - f. The location of the breakers that will be opened, and the places from which they can be controlled.
  - g. Describe the position indication and status signals which will be lost as a result of deenergizing these components.
  - h. Describe in detail, the administrative procedures which will be used to assure proper equipment alignment and the supervisory personnel responsible for maintaining control.
  - i. Indicate the RCS temperature/pressure conditions for which the accumulator isolation valve will be closed, and describe the location of the breaker, and the places from which it may be controlled.

- j. Describe the impact on overall plant operations if you routinely lowered accumulator nitrogen pressure when in a cold shutdown condition.
4. The staff has noted that several Appendix G violations have occurred during component or system tests while in cold and shutdown conditions. In this regard, please address the following questions.
    - a. What components or systems that could cause overpressure transients, are routinely tested while in a cold shutdown condition?
    - b. What extra measures are taken to prevent an overpressure event during these tests?
  5. The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operators's attention to a transient in progress.

POSITION: The staff will require that if it is not presently installed, such an alarm must be installed as soon as possible. Accordingly, please furnish the following information:

- a. Your method to provide the alarm, and the associated time schedule, or your justification for why this cannot be done.
  - b. A synopsis of the system modifications that are necessary.
  - c. The alarm setpoint, mode of annunciation and sensor.
  - d. How you ensure that the alarm is available and operating properly during all water-solid operations and how you minimize its down-time for all other cold shutdown conditions.
6. The RHR (or SCS) is normally connected to the RCS and operating when the plant is in a cold shutdown condition. The inadvertent isolation of the RHR system while water solid has caused a number of overpressure transients, and the RHR safety valve has actually terminated others. The RHR (or SCS) therefore plays an important part in the initiation and possible mitigation of the PWR overpressurizations. Accordingly, we request the following additional information:

- a. RHR (or SCS) design pressure.
  - b. A description of the system isolation valves and their arrangement (e.g., number and configuration of valves installed, and pneumatic or motor operated).
  - c. Interlocks, interlock setpoints, and alarms associated with each isolation valve.
  - d. Nominal stroke time of isolation valves.
  - e. The setpoint and capacity of RHR (or SCS) relief and safety valves.
  - f. All pressure alarms, setpoints and associated annunciation for the system.
7. Reactor coolant system heatups, resulting from improper operation of the reactor coolant pump (RCP) while in a cold, shutdown and water solid condition, have been responsible for approximately 15% of the RCS overpressurization events.

POSITION: We will require that all licensees include adequate provisions to prevent RCP starts while in a water solid condition unless such starts are absolutely necessary. In those cases where the RCP starts cannot be avoided, the licensee should take appropriate steps to determine and minimize the RCS temperature profile.

Based on the position stated above, provide the following information.

- a. Describe the normal operating conditions during which the RCS is maintained water solid with all RCP's stopped (e.g. fill and vent, pressurizer cooldown).
- b. For each of the above procedures, justify your inability to establish a N<sub>2</sub>, air or steam bubble in the pressurizer prior to the start of the first RCP.
- c. What are the limits associated with system temperatures before the first RCP can be started in a solid RCS?



- d. Specify the instruments utilized to determine the RCS temperature profile.
- e. Provide the necessary schematics and procedural descriptions that show what your actions would be to bring the RCS to an isothermal condition.
- f. Summarize any other measures you take to reduce possible RCS pressure spikes during RCP starts, (e.g. open all letdown orifice isolation valves, stop makeup flow, etc.).



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

November 17, 1976

DOCKET NOS.: 50-344, 50-213, 50-315, 50-244, 50-247, 50-286, 50-305,  
50-266, 50-301, 50-282, 50-306, 50-261, 50-295, 50-304,  
50-206, 50-280, 50-281, 50-250, 50-251, and 50-334.

LICENSEE/FACILITY:

ROCHESTER GAS & ELECTRIC CORPORATION (R. E. GINNA)  
CONSOLIDATED EDISON COMPANY (INDIAN POINT UNITS 2 & 3)  
DUQUESNE LIGHT COMPANY (BEAVER VALLEY UNIT 1)  
CONNECTICUT YANKEE ATOMIC POWER COMPANY (HADDAM NECK)  
WISCONSIN PUBLIC SERVICE CORPORATION (KEWAUNEE)  
WISCONSIN ELECTRIC POWER COMPANY (POINT BEACH UNITS 1 & 2)  
NORTHERN STATES POWER COMPANY (PRAIRIE ISLAND UNITS 1 & 2)  
CAROLINA POWER LIGHT COMPANY (H.B. ROBINSON)  
SOUTHERN CALIFORNIA EDISON COMPANY (SAN ONOFRE)  
VIRGINIA ELECTRIC & POWER COMPANY (SURRY UNITS 1 & 2)  
PORTLAND GENERAL ELECTRIC COMPANY (TROJAN)  
FLORIDA POWER & LIGHT COMPANY (TURKEY POINT UNITS 1 & 2)  
COMMONWEALTH EDISON COMPANY (ZION UNITS 1 & 2)  
INDIANA & MICHIGAN POWER COMPANY (D.C. COOK, UNIT 1)

SUMMARY OF MEETING HELD ON NOVEMBER 4, 1976, CONCERNING PROPOSED MEASURES TO PREVENT REACTOR VESSEL OVERPRESSURIZATION IN OPERATING WESTINGHOUSE (PWR) FACILITIES.

On November 4, 1976, we met with representatives of PWR licensees with Westinghouse designed plants to discuss measures being taken to prevent reactor vessel overpressurization.

A list of attendees is attached.

Significant discussions are summarized below.

We summarized the correspondence and discussions that have occurred with the Westinghouse licensees since our generic letter on reactor vessel overpressurization was issued in August 1976. We then identified the criteria listed below as those that should be applied in the design of equipment which is to prevent pressure transients that would exceed the limits of Appendix G to 10 CFR §50.

~~PAR 8111 050085~~

1. Credit of Operator Action - No credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress.
2. Single Failure Criteria - The pressure protection system should be designed to protect the vessel given a single failure in addition to a failure that initiated the pressure transient. In this area, redundant or diverse pressure protection systems would be considered as meeting the single failure criteria.
3. Testability - The equipment design should include some provision for testing on a schedule consistent with the frequency that the system is used for pressure protection.
4. Seismic Design and IEEE 279 Criteria - Ideally, the pressure protection system should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

Representatives of the task group of Westinghouse utilities formed to evaluate the problem of reactor vessel overpressurization provided a description of the steps they have taken to respond to the requirements set forth in our August 1976 letter. A summary was given of the various types of Thermal and mass input transients being considered, and it was indicated that a "bounding" analysis is being performed to consider the worst case situation for all Westinghouse plants. The preliminary results of the mass input analysis show that the pressurizer power relief valves have both the capacity and time response characteristics to limit the resultant pressure surges. The task group, however, indicated that a more detailed analysis would be necessary in the case of the pump-start or thermal type of transient before any similar determination could be made. The detailed plant specific analyses are not scheduled for completion for about six months. Since the power operated relief has evidently been selected by the licensees as the means to limit pressure transients, we urged that efforts be made to begin ordering the necessary equipment now rather than waiting 6 months for the plant specific analyses results. We also urged that the licensees concurrently investigate other factors such that installation times can be minimized. The licensees' task group agreed to look into these matters.

We indicated that the "single failure" criteria being assumed by the licensees was not consistent with the conventional single failure criteria required by the staff. The licensee agreed to examine this area further and to provide justification for any deviations from the conventional single failure criteria. This information as well as a discussion of the various administrative measures that the licensees intend to use to prevent pressure transients while shutdown are to be prepared with a target date of submittal to the staff by December 3, 1976.



Gary G. Zech, Project Manager  
Operating Reactors Branch #1  
Division of Operating Reactors

Attachment:  
List of Attendees

Meeting Summary for  
W plants

- 4 - November 17, 1976

Docket File  
NRC PDR  
LOCAL PDR  
ORB#1 Reading  
NRR Reading  
B. C. Rusche  
E. G. Case  
V. Stello  
K. R. Goller  
D. Eisenhut  
T. J. Carter  
A. Schwencer  
D. Ziemann  
G. Lear  
R. Reid  
R. Clark  
L. Shao  
R. Baer  
W. Butler  
B. Grimes  
Project Manager  
Attorney, OELD  
OI&E (3)  
S. M. Sheppard  
Participants (NRC)  
R. Fraley, ACRS (16)  
T. B. Abernathy  
J. R. Buchanan

NRC STAFF MEETING WITH WESTINGHOUSE PWR LICENSEES

NOVEMBER 4, 1976

ATTENDANCE LIST

NRC

G.G.Zech  
R.L.Baer  
G.Lanik  
L.B.Marsh  
J.E.Ouzts  
T.J.Carter  
I.Villalva  
W.A.Paulson  
S. Israel  
J.Mazetis  
D.Tibbitts  
D. Neighbors  
W.T.Russell  
M.B.Fairtile  
R.J.Cook  
P.B.Erickson  
P.E.Harmon  
J.Stosnider  
R.M.Gamble  
R.Clememson  
M.Grotenhuis  
D.Hood  
W.Pike  
R.W.Klecker  
J.Wetmore  
M.Fletcher  
T.Wambach

CAROLINA POWER & LIGHT  
CO.

D.B.Waters  
R.G.Black

SOUTHERN CALIFORNIA EDISON

W.C.Moody

NUSCO

B.Ilberman  
M.Kupinski  
P.F.Santoro

CONNECTICUT YANKEE ATOMIC  
POWER CO.

J.Levine

DUKE POWER CO.

R. W. Revels  
E. M. Geddie, Jr.

VIRGINIA ELECTRIC & POWER CO.

D. W. Speidell, Jr.  
A. L. Hogg, Jr.

ALABAMA POWER CO.

J. R. Campbell

COMMONWEALTH EDISON CO.

T.R.Tramm  
E.E.O'Brien  
W.A.Wogsland

CONSOLIDATED EDISON CO.

C.W.Jackson  
P.M.Pivawer  
J.Makepeace

POWER AUTHORITY OF THE  
STATE OF NEW YORK

P.F.Altern  
J.M.Vargas

WISCONSIN ELECTRIC POWER CO.

R. A. Newton

WISCONSIN PUBLIC SERVICE CORP

M.E.STERN

ROCHESTER GAS & ELCTRIC

R.W.Elias  
B.A.Snow

AMERICAN ELECTRIC POWER

P.W. Daley  
R.L. Shoberg  
J.G. Dell Perico

DQUESNE LIGHT COMPANY

S. R. Porter  
J. J. Carey

Attendance List

- 2 -

FLORIDA POWER & LIGHT

C.S.Pillar  
M.A.Schopmann

WESTINGHOUSE - PWRSD

W. G. Poulson  
F. Gilgliotti  
H. Gutzman  
A. M. Sklencar  
R. W. Fleming  
R. C. Jenkner

NOTHERN STATES POWER CO.

L. L. Taylor

PUBLIC SERVICE E&G CO.

L.K.Miller

SOUTHERN COMPANY SERVICES, INC.

J. R. Lyons

PORTLAND GENERAL ELECTRIC CO.

D. I. Herbon

SNUPPS

F. Schwoerer



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

OCTOBER 23, 1989

MEMO TO DOCUMENT CONTROL

FROM : G. REQUA PM, H.B. ROBINSON,  
URB-1

SUBJECT: IMMEDIATE BACKFIT (DOCKETING)  
NRC LETTER DATED JANUARY  
10, 1977 & ENCLOSURES

DUE TO FOIA - 84-790 CONCERNING  
THE SUBJECT DOCUMENT, I AM  
REQUESTING THAT THE DOCUMENT & ENCL'S  
BE BACKFITTED AND PLACED IN THE  
PUBLIC DOCUMENT ROOM. A COPY OF  
THE DOCUMENT & ENCLOSURE IS ATTACHED  
FOR YOUR INFO. THIS REQUEST HAS  
BEEN DISCUSSED WITH MIKE COLLINS



