04/18/78

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)
DISTRIBUTION FOR INCOMING MATERIAL 50-261

REC: SCHWENCER A
NRC

ORG: UTLEY E E
CAROLINA PWR & LIGHT

DOCDATE: 04/11/78
DATE RCVD: 04/17/78

DOCTYPE: LETTER NOTARIZED: NO

COPIES RECEIVED
LTR 1 ENCL 40

SUBJECT:
RESPONSE TO NRC REQUEST OF 03/10/78... FORWARDING ADDL INFO CONCERNING THE 01/29/78 OVERPRESSURIZATION OCCURRENCE AT UNIT 2... W/ATT SUPPORTING AND RELATING INFO.

PLANT NAME: H B ROBINSON - UNIT 2

REVIEWER INITIAL: XJM
DISTRIBUTOR INITIAL: w

****************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS ******************

REACTOR VESSEL OVERPRESSURIZATION DISTRIBUTION PER G. ZECH 10/21/76.
(DISTRIBUTION CODE A010)

FOR ACTION: BR ~~CHIEF SCHWENCER~~**W/7 ENCL

INTERNAL:
REG FILE**W/ENCL
~~I & E**W/2 ENCL~~
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BAER**W/ENCL
ZECH**W/ENCL
NRC PDR**W/ENCL
OELD**LTR ONLY
BOSNAK**W/ENCL
NOVAK**W/ENCL
SHAO**W/ENCL
BUTLER**W/ENCL

EXTERNAL:
LPDR'S
HARTSVILLE, SC**W/ENCL
TIC**W/ENCL
NSIC**W/ENCL
ACRS CAT B**W/16 ENCL

DISTRIBUTION: LTR 40 ENCL 38
SIZE: 1P+8P+1P

CONTROL NBR: 781070045

************************************ THE END ************************************

CP&L
Carolina Power & Light Company

April 11, 1978

REGULATORY DOCKET FILE COPY

FILE: NG 3514 (R)

SERIAL: GD 78 1061

1978 APR 17 AM 10 55

RECEIVED DISTRIBUTION SERVICES UNIT

Office of Nuclear Reactor Regulation
ATTN: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
OVERPRESSURE PROTECTION - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Dear Mr. Schwencer:

By letter of March 10, 1978, you requested additional information concerning the January 29, 1978 overpressurization occurrence at the H. B. Robinson Unit No. 2. Enclosure 1 to this letter is in response to your request.

In addition, a re-review was requested of all plant procedures. This review was requested to ensure that low reactor coolant system temperature pressure transients that would challenge the Overpressurization Protection System are avoided. This review is included as a part of Enclosure 1.

Please advise us if you have any questions regarding the attached information.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

JMC/CSB/gsm
Attachment

cc: Mr. James P. O'Reilly

781070045

A010 5 1/40*

Enclosure 1

A. In accordance with your request of March 10, 1978, a re-review of the operating manual has been completed. During performance of the review, consideration was given to the recommendations per parts 1 and 4 of the Staff's request for additional information. The following is a summary of the review.

I. Volume 1 - Administrative Instructions

1. No change issued. This volume contains no impacting procedures.

II. Volume 2 - System Descriptions

1. No change issued. The volume contains no impacting procedures.

III. Volume 3 - Operating Procedures

1. Procedure OP-23, Reactor Protection, was changed to add an initial condition concerning the operability of the overpressure protection system.
2. Procedure OP-25, Reactor Coolant System, was revised to reflect the new heatup and cooldown limitations imposed by Technical Specifications. The valve alignment procedure for the over-pressure protection system was incorporated into this procedure.
3. Procedure OP-28, Charging and Volume Control, was revised to include an initial condition lowering the allowable maximum pressure for solid operation on RHR. A precaution was added to avoid overpressure transients during operation.
4. Procedure OP-30, Pressurizer Pressure and Spray, was completly revised to reflect the new overpressure protection system. Revised limitations concerning heatup and cooldown curves were added to comply with the Technical Specifications. Precautions were added at appropriate steps to avoid overpressure transients.
5. Procedures 38 and 38B, RHR System, were revised to reflect new maximum operating pressures when solid. Precautions were added to avoid isolation of letdown when solid and on RHR. A step was added to ensure that the overpressure protection system is operable.
6. Procedure OP-50, RCS Filling and Venting, was revised to delete an unused procedure for cold hydrostatic testing of the RCS. It was originally used during initial startup and never deleted once new Appendix G curves were issued. Steps were added to ensure the operability of the overpressure protection system prior to filling the RCS. Precautions were added to ensure that the operator is alert to potential overpressure situations.

IV. Volume 4 - Overall Plant Operating Procedures

1. Procedure GP-1A, Plant Heatup, was revised to reflect new heatup and cooldown limitations, to ensure the operability of the over-pressure protection system during water solid conditions, and to add precautions to alert the operators to potential over-pressure situations.
2. Procedure GP-1D, Plant Cooldown, similar changes were initated as were to GP-1A.

V. Volume 5 - Abnormal Operating Procedures

1. No changes were required.

VI. Volume 6 - Emergency Instructions

1. No changes were required.

VII. Volume 7 Precaution, Limitations, and Setpoints

1. Procedure PLS-2, Reactor Coolant System, was revised to reflect the new heatup and cooldown rates for the pressurizer.

VIII. Volume 8 - Radiation Control and Protection

1. No changes required.

IX. Volume 9 - Refueling Manual

1. No change required.

X. Volume 10 - Periodic Tests

1. Procedure PT-2.7 A, B and C, Safety Injection Components, were revised to avoid an overpressure occurrence when water solid.
2. Procedures PT-2.1 and PT-23.2, Safety Injection and Site Blackout Tests, will be revised to avoid an overpressure transient by ensuring the RCS is at cold shutdown and in the non-water solid condition prior to testing. This procedure will be revised prior to their next performance.
3. Procedure PT-2.8, RHR Component Test, this test will be revised or a new test developed for use at cold, water solid conditions. It is currently incompatible with those conditions. This is a new test developed for the ISI program and is suitable only for performance with the RHR System in standby.

XI. Volume 11 - Quality Assurance

1. No change required.

XII. Volume 12 - Maintenance Instructions

1. No change required.

XIII. Volume 13 - Emergency Plan

1. No change required.

XIV. Volume 14 - Special Nuclear Material Accountability

1. No change required.

XV. Volume 15 - Curve Book

1. No change required.

XVI. Annunciator Procedure

1. Procedure A-1, Miscellaneous NSSS Annunciators, was revised to reflect the necessary action to be taken should one of the two overpressure protection system annunciators energize to the alarm condition.

XVII. Industrial Security

1. No change required.

XVIII. Fuel Follow Procedure

1. No change required.

XIX. Fire Protection Procedures

1. No change required.

With the changes noted and the complete re-review described, appropriate and sufficient instructions are contained in or are being added to plant procedures which will reasonably assure that operator caused low temperature RCS pressure transients which could challenge the overpressure protection system are avoided.

B. The following items address your request for specific additional information:

1. Provide justification of the bases and prerequisite steps of SIS tests, PT 2.1 and 23.2 which require performance of these tests while the plant is in a water-solid condition. It is our position that these tests can be performed while the Reactor Coolant System (RCS) is other than in a water-solid condition.

   Response

   It has been determined that these tests can be performed with a non-water solid system while at cold shutdown.

   The subject periodic tests are therefore being revised such that the prerequisites will require the plant to be at cold shutdown and the Reactor Coolant System be in the non-water solid condition.

2. You have proposed to determine the SG/RCS $\Delta T$ input by measuring the steam generator secondary side water temperature using a hand held pyrometer at a location about four feet above the tube sheet. This location could be about 30 feet below the elevation of maximum water temperature. Justify that the proposed location for taking measurements is adequate by showing that the SG/RCS $\Delta T$ is conservative when taking into account measurement activities.

   Response

   The Westinghouse Electric Corporation has determined (letter attached) that shell and bulk water temperature would reach an isothermal condition a short period after securing feedwater flow following shutdown. With this isothermal condition a temperature measurement obtained on the steam generator shell above the tube would represent bulk water temperature regardless of the height of measurement. The temperature measurement, taken for RCS startup, would be performed a considerable time after shutdown.

3. You have documented your thorough investigation of the recent over-pressure event, including the sequence of events and operator actions. Provide additional analyses to substantiate the postulated causes and resulting RCS transient behavior including an explanation for the termination of the event. Assume a best estimate of the plant parameters at the time of the transient, including the heat input from decay heat and the two RCP's which were operating at the time, and other pertinent plant data such as RHR heat exchanger flow rates and $\Delta T$ to support their analyses, giving attention to RHR safety valve setpoints and RHRS pressure rise on initiation of RHR pumps.

Response

The results of the initial investigation of the January 29, overpressurization occurrence was provided in the Supplemental Information submitted with Licensee Event Report LER-78-3. This report was submitted to Mr. James P. O'Reilly by letter of February 13, 1978 (CPL-Serial No. GD-78-404).

The information submitted with LER-78-3 details the results of the investigation made and presents the best estimate available of plant operating conditions and parameters existing before, during, and after the occurrence. It was shown that the actual pressure experienced during the occurrence was 530 PSIG. It was shown that the increase in pressure after shutting down the Residual Heat Removal (RHR) Pumps was due to the heat input to the solid system caused by decay heat and heat from running the Reactor Coolant Pumps. The calculated rise in pressure due to the combination of mass input from the running charging pump and the heat input from decay and RCPs was much greater than the actual rise experienced. The actual pressure rise experienced was less than either of these. It was thus concluded by these calculations that letdown flow was maintained.

The analyses and evaluations presented in the LER submittal were based on an indepth review of all available information recorded during the occurrence and post-occurrence reviews with personnel involved in the performance of the refueling Periodic Test. The analyses and results presented identify as accurately as possible the events leading up to and resulting in the occurrence as well as the action taken to limit the magnitude and end the occurrence.

Additional analyses concerning the method and events occuring during termination of the occurrence were requested. Another review of the events was conducted with specific attention being placed on the pressure seen by the RHR loop and its pressure relief valve (RHR-706) in an effort to determine if the lifting of the relief valve caused termination and reversal of the pressure rise.

The additional review resulted in the same results with regard to the peak pressure experienced, and the termination of the occurrence was again concluded to be the initiation of the RHR flow as a result of the manual initiation of the Safety Injection Signal as required by the periodic test being performed.

In the additional review, analyses and evaluations were made to determine if the relief valve (RHR-706) lifted to provide a relief path for the RHR system and thus the RCS when the RHR pumps were initiated. No physical proof exists that would indicate that the valve did or did not relieve.

The maximum pressure which the relief valve could have experienced during the occurrence was investigated. During recent operation of the RHR system when heating up the RCS, the differential pressure was determined across the pump. A value of 105 psid was obtained. Ignoring head and frictional losses in pipes and valves and pressure drops across the heat exchangers, a maximum

pressure of 635 psig could have been experienced during the Jan. 29, occurrence. However, this maximum pressure is not realistic. By determination of head losses due to height differential between the pump and the relief valve, a maximum pressure of 620 psig would have been experienced at the valve. Analytically it is possible to determine a pressure drop across the heat exchangers and the frictional losses resulting from flow through the pipes and valves. However, it would not be practical since valve positions and individual heat exchanger flow rates were not recorded.

Any amount of pressure drop due to flow through the heat exchangers and frictional losses in the pipe and valves should result in the pressure experienced by the relief valve of less than 620 psig.

The relief valve setpoint is 600 $\pm$ 18 psig. Thus the pressure experienced could have caused the relief valve to lift. The minimum pressure at which the valve would have reseated according to valve specifications could have been approximately 540 $\pm$ 18 psig. However, the pressure dropped to well below 500 psig with apparently a constant slope as was graphically depicted in the attachment to LER-78-3.

During the review, records of the 1976 refueling were examined. It was determined from this review that a similar pressure rise associated with the performance of PT 2.1 occurred during the 1976 refueling. The similarity was in the magnitude of the rates of pressure increase and decrease and duration of transient. The startup pressures and finishing pressures were both lower than during the 1978 occurrence and no violation to the Technical Specification limits occurred. Peak pressure during the 1976 P.T. was 400 psig with a starting pressure of 250 psig. Under the operating conditions existing during the 1976 performance of the periodic test, the peak pressure after the pumps restarted at that time would not have resulted in pressures in the RHR loop sufficient to cause the RHR system relief valve to lift. Therefore, the RHR system is capable of terminating pressure rises and decreasing pressure at a rate consistent with that experienced in 1978 without the relief valve lifting.

The description, analyses, and results presented in LER-78-3 are not altered by our additional review. Regardless of whether or not the RHR relief valve lifted, it was the initiation of the RHR pumps which caused termination and reversal of the pressure rise. The conclusions reached in the LER-78-3 remain viable.

4. Based on our review of the overpressure event at Robinson, we recommend that the following procedural changes be made or that justification be provided why these changes are not necessary:

   a. When operating with-solid RCS, any adjustments in RHR flow (RHR pump start/stop or heat exchanger bypass flow changes) should be effected with careful monitoring and control of the letdown flow control valve (RHR to CVCS) to avoid aby significant RCS pressure transients.

   b. When the RCS is water-solid and RCS pressure is being controlled with the RHR system letdown flow path, the three block orifice

discharge valves in the CVCS should remain fully open to help dampen any RCS pressure fluctuations.

c. When operating on the RHR system while the RCS is water-solid, stop all unnecessary RCP's before stopping the RHR pump to minimize the thermal input to the system before the heat removal capacity is removed.

d. During RCS cooldowns, operate at least one RCP until the RCS temperature is at or below 150$^{\circ}$F to minimize potential temperature assymetries during subsequent RCP startups.

Response

a. The precaution is being incorporated into applicable plant procedures.

b. This recommendation is not being incorporated into plant procedures. During RHR operation, the letdown line from the RHR system through HCV-142 to the CVC system is the normal purification path. The RHR system letdown pressure through HCV-142 is at a higher pressure than RCS pressure letdown through the three-block orifice discharge valves. Both letdown lines become a common line from the containment isolation valves CVC 204A and CVC 204B through the CVC system. The design of the systems is thus incompatable for simultaneous operation. A flow path would be established which would effectively bypass purification in this configuration. Flow from the RHR letdown line would be forced back through the letdown orifices and thus back into the RCS in the reverse direction from normal letdown flow. To adjust RHR letdown pressure to avoid such an occurrence would present a configuration which would be subject to pressure fluctuations which it is intended to prevent.

RHR letdown is controlled by HCV-142 and PCV-145. If HCV-142 was appropriately adjusted to prevent back flow through the orifices, PCV-145 would still control letdown. However, CVC-204A and CVC-204B downstream from the orifices and upstream from HCV-142 and PCV-145 receive Phase A isolation. Thus any SI signal or loss of power to those valves would result in the closing of the valves and creation of an undersirable pressure fluctuation.

With the changes adopted as a result of comments 1 and 3 of the request for additional information, the results of our reviews of plant procedures, and the problems presented above, this change is neither desirable nor necessary.

c. The proposed change is being incorporated into appropriate plant procedures.

d. The recommended change will not be made. It is not necessary to operate a RCP until RCS pressure is at or below 150$^{\circ}$F to minimize potential temperature assymetries during subsequent RCP startups.

Procedures exist which limit the operation and startup of Reactor Coolant Pumps with regard to system pressure and temperature. In addition, such a procedure would be overly restrictive. At cold shutdown, the RCS temperature is less than 200°F. A procedure requiring RCP operation down to 150°F is not desirable since there would be cases when RCS temperature is not required to be below 150°F, yet RCP operation would not be required or beneficial. With existing procedures and the over restrictiveness of such a procedure it is determined that this change is not necessary.

Westinghouse
Electric Corporation

Box 4808
1299 Northside Drive NW
Atlanta Georgia 30302

CPL-78-513

March 28, 1978

Mr. R. B. Starkey, Plant Manager
Carolina Power & Light Company
H. B. Robinson SEG Plant
P. O. Box 790
Hartsville, South Carolina 29550

Dear Mr. Starkey:

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON UNIT 2
STEAM GENERATOR BULK WATER TEMPERATURE

In answer to the question; How long after securing feedwater flow to the Steam Generator under the following conditions does the Steam Generator shell temperature represent bulk water median temperature?

Conditions

1. Unit cooldowned from operating temperature to cold shutdown.
2. Feedwater to Steam Generator secured.
3. RCS flow secure. (RCS temperature approximately 120° F)

In Ⓦ judgement, based on the design thermal analysis of Steam Generators, the Steam Generator shell and bulk water would reach an isothermal condition between three and four hours after securing feedwater flow. This takes into consideration that the insulation on the steam generator keeps shell temperature gradient to a minimum and therefore the unit will cooldown as a isothermal unit. If the temperature was measured on the steam generator shell between tube sheet and water level this would represent median bulk water temperature.

If you have any further questions please contact me.

Very truly yours,

R. J. Muth

R. J. Muth
Customer Service Engineer
Southern Service Region

cc: H. R. Banks
B. J. Furr
R. M. Coats
D. B. Waters
J. F. Halifax
J. M. Curley

Standard Form 1047
September 1973
4 Treasury FPM 2000
1047-106

# PUBLIC VOUCHER FOR REFUNDS

Voucher No. File

Schedule No. C0886-01

50-261

U. S. NUCLEAR REGULATORY COMMISSION
(Department or Establishment, Bureau or Office)

Location: WASHINGTON, D.C. 20555

Appropriation or Fund: 31X0200

PAID BY

4-11-78

To Address
CAROLINA POWER & LIGHT COMPANY
ATTN: MR. M.A. MCDUFFIE, SENIOR VICE
PRESIDENT ENGNRNG & CONSTRUCTION
336 FAYETTEVILLE STREET
P. O. BOX 1551
RALEIGH, NORTH CAROLINA 27602

Deposit received from the above-named depositor on ____, 19__

for ____

has been applied as herein stated and the balance indicated is returned herewith:

Amount of deposit ____ $____

Applied as explained in "Remarks" below ____ ____

____ ____

Balance authorized to be refunded ____ $ 57,200.00

Remarks:

REFUND OF ANNUAL OPERATING FEES - DPR-23

H. B. Robinson 2

(Sign original only) ____

Title ____

Refund by
Check No. ____
Cash, $____ on ____ (Signature of payee) ____
(Sign original only)
Other method, $____, ____
(Describe)

Carolina Power & Light Company

April 3, 1978

FILE: NG 3514 (R)

SERIAL: GD-78-962

Office of the Controller
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON UNIT NO. 2
OPERATING LICENSE NO. DPR-23
REFUND OF ANNUAL OPERATING FEES

Dear Sir:

Pursuant to your notice in the February 21, 1978 Federal Register, Carolina Power & Light Company hereby requests a refund of annual fees which we paid prior to the suspension of such fees by the Commission in 1974. The following is a list of the three annual fees which we believe are eligible to be refunded to the Company. These fees were paid for Operating License DPR-23 for our H. B. Robinson Steam Electric Plant Unit 2 for the periods July 31, 1971 - July 30, 1972, July 31, 1972 - July 30, 1973, and July 31, 1973 - July 30, 1974.

| Commission Invoice No. | Date of Invoice | Date Paid | Refund Requested | Previous Refund Requested |
|---|---|---|---|---|
| Invoice Number 99-72 | 7-14-71 | 7-30-71 | $ 4,400 | 0 |
| Invoice Number 146-73 | 8-1-72 | 8-15-72 | 26,400 | 0 |
| Invoice Number 64-74 | 7-9-73 | 7-23-73 | 26,400 | 0 |
| | | Total | $ 57,200 | 0 |

We appreciate your attention to this matter in supplying the requested refund.

Yours very truly,

M. A. McDuffie

M. A. McDuffie
Senior Vice President
Engineering & Construction

DLB/gsm

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

APRIL 10 1978

DISTRIBUTION
Docket
NRC PDR
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ASchwencer
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OELD
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JRBuchanan
DEisenhut

Docket No. 50-261

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

Gentlemen:

On May 18, 1977, you responded to our letter dated March 28, 1977, which requested information regarding the Robinson 2 diesel generators. We have reviewed your response and have the following comments and positions with respect to corrective measures:

1. In response A.(1), you indicate that "Energization of the shutdown relay" condition shares the "Emergency D/G Trouble" window. We do not consider this acceptable. You should provide a separate alarm for each disabling condition or a single shared alarm (with reflash capability) for all conditions with wording clearly indicating that the D/G is incapable of an automatic start.

2. The following conditions:

   a. Local or remote control switches (2) not in automatic position.

   b. D/G breaker trip lockout relay not reset.

   c. Local manual D/G fuel rack mechanical lockout not reset.

   should be alarmed as discussed in (1) above.

You should respond to this letter within 60 days and describe any modifications that you propose for achieving compliance with the staff positions stated above or justify alternative actions.

Sincerely,

/S/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ➤ | DOR:ORB#1 | DOR:ORB#1 | | | | |
| SURNAME ➤ | DNeighbors:lb | ASchwencer | | | | |
| DATE ➤ | 4/10/78 | 4/10/78 | | | | |

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

APRIL 10 1978

Docket No. 50-261

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

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Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D.C. 20036

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