Distribution
Docket File
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Attorney, OELD
IE (3)
N. Dube(w/o Tech specs)
M. Jinks(2/encl per docket)
R. C. DeYoung
A. Braitman, OAI(w/o tech specs)
R. Powell
M. Rushbrook
S. Kari(w/o tech specs)
W. Miller, EDO:AD(w/o tech specs)
EP Project Manager
LWR 1 BC's(w/o tech specs)
ACRS (16)
B. Scharf (15 cys)
A. Steen

R. Klecker
M. Williams
K. Goller
D. Skovholt
E. Hughes(No admendmts

AUG 1 1 1975

Docket No. 50-324

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

Amendment No. 3
Change No. 3
License No. DPR-62

Gentlemen:

By your letter, dated April 3, 1975, you transmitted a proposed change to the Technical Specifications of License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2. This change defines a new temperature limit for the suppression pool water to provide additional assurance of maintaining primary containment integrity and function in the event of extended relief valve operation.

A notice of proposed issuance of amendment for this action was published in the Federal Register on June 19, 1975 giving members of the public an opportunity to request a hearing if their interests were affected by this action. The time for requesting a hearing in the form of a petition for leave to intervene expired at midnight on July 21, 1975. To date, we have not received any petition for leave to intervene.

On July 9, 1975, the Office of Nuclear Reactor Regulation issued an 'Order Modifying License and Revoking Order to Show Cause.' This Order added a condition to Operating License No. DPR No. 62. We have included this condition along with the technical specification change as Amendment No. 3 to DPR-62.

We have enclosed for your information and use Amendment No. 3 to License DPR-62, the supporting Staff Evaluation, a Federal Register Notice which

bcc: J. R. Buchanan, ORNL
Thomas, B. Abernathy, DTIE
A. Rosenthal, ASLAB
N. H. Goodrich, ASLBP

OFFICE➤						
SURNAME➤						
DATE➤						

has been forwarded to the Office of the Federal Register for publication and Change No. 3 to the Technical Specifications.

Sincerely,
Original Signed by
W. R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3 - DPR-62
2. Staff Evaluation
3. Federal Register Notice
4. Change No. 3 - Technical Specifications

cc: Richard E. Jones, Esq.
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Mr. W. A. Kopp, Jr.
Chairman, Board of County
Commissioners of Brunswick County
Bolivia, North Carolina 28422

OFFICE ➤	L:LWR 1-2	L:LWR 1-2	OELD	L:LWR 1-2		
SURNAME ➤	MRushbrook/red	RPowell	STRIDIRON	WRButler		
DATE ➤	7791 8/4/75	8/6/75	8/1/75	8/1/75		

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

CAROLINA POWER AND LIGHT COMPANY
DOCKET NO. 50-324
BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3 Change No. 3
License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated April 3, 1975 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by the changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2C(2) of Facility License No. DPR-62 is hereby amended to read as follows:

"2C(2) Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 3."

This license is further amended by adding the following condition:

"2C(3) Carolina Power and Light Company will undertake a program for seismic monitoring for a minimum of two years unless termination is earlier approved by the NRC staff. The program and its control will be conducted in general conformity with the document "Brunswick Steam Electric Plant Program for Seismic Monitoring" dated June 10, 1975 as revised June 27, 1975 and attached* hereto as Appendix A. The program will include: 1) not less than ten seismic monitoring stations (seven permanent and three portable), in an array approved by the NRC staff, unless a lesser number is approved by the NRC staff in writing, and 2) quarterly reports on the monitoring data to be submitted to the NRC. Should the NRC staff determine that initiation of Phase II as described within the program within the two year monitoring period, or Phase III following initiation of Phase II, is required the licensee will either comply with a request to proceed to Phase II (or Phase III) or immediately request and be granted a hearing on the issue of whether the data on which the staff's request is based justifies the initiation of Phase II (or Phase III) under the program for seismic monitoring agreed to by the licensee and the NRC staff. Nothing herein will be construed as precluding changes in the program by the licensee which do not adversely affect the quantity of information derived from the monitoring program. NRC will be informed of any such changes in the quarterly report.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

(for) R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Division of Reactor Licensing

Date of Issuance: AUG 1 1 1975

*Appendix A is not provided in the license but is available for public inspection in the NRC's public document room at 1717 H Street, N.W., Washington, D. C. and at the Southport Brunswick County Library, 109 W Moore Street, Southport, North Carolina.

ENCLOSURE 2

CHANGE NO. 3

SAFETY EVALUATION SUPPORTING
CHANGE TO TECHNICAL SPECIFICATIONS
FOR BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2 (DPR-62)

Introduction

By letter dated April 3, 1975, the licensee, Carolina Power & Light Company requested a change in the Technical Specifications appended to Operating License No. DPR-62 for the Brunswick Steam Electric Plant, Unit 2 located at Southport, North Carolina. The proposed change in Technical Specifications was submitted in response to our request to the licensee dated February 18, 1975 and is responsive to the guidelines set forth in our letter. We have made additional modifications to these proposed Technical Specifications to improve the clarity and intent of the specification and its basis. The proposed change in Technical Specifications defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment function and integrity in the event of extended relief valve operation.

Discussion

The Brunswick Plant, Unit 2 is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

1. Steam Vent Clearing Phenomenon

With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 18, 1975 we also requested each applicable licensee to provide information to demonstrate that the torus structure will maintain its integrity throughout the anticipated life of the facility. Because of apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

2. Steam Quenching Vibration Phenomenon

The steam quenching vibration pehnomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads in the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself, due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event. Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin[1/] exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

Evaluation

The existing Technical Specifications for Brunswick Unit 2 limit the torus pool temperature to 95°F. This temperature limit assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain well below 95°F. While this 95°F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications.

This action was, as discussed in our February 18, 1975 letter, first suggested by the General Electric Company (GE) who had earlier informed us of the steam

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

quenching vibration occurrences at a meeting on November 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in those communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon. Our implementation of the GE recommended procedures and temperature limits via changes in the Technical Specifications are evaluated in the following paragraphs:

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This new limit and associated requirement to scram the reactor provides additional margin below the 170F temperature related to potential damage to the torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit applicable to surveillance testing of relief valves and RCIC or HPCI operation provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications is a maximum suppression pool water temperature of 120°F.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

d. In addition to the new limits on temperature of the torus pool water, discussion in the Bases includes a summary of operator actions to be taken in the event of a relief valve malfunction. These operating actions are taken in order to avoid the development of temperatures approaching the 170F threshold for potential damage by the steam quenching phenomenon.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: JUN 1 8 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-324

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-62, which was issued to Carolina Power & Light Company on December 27, 1974. Amendment No. 3 to DPR-62 revises the Technical Specifications for operation of the Brunswick Steam Electric Plant, Unit 2, and incorporates into the license Condition 2.c.(3) authorized by Ben C. Rusche, Director, Office of Nuclear Reactor Regulation in an "Order Modifying License and Revoking Order to Show Cause", dated July 9, 1975. The Brunswick site is on the Cape Fear River, near Southport in Brunswick County, North Carolina. The amendment is effective as of its date of issuance.

The Technical Specification change defines a new temperature limit for the suppression pool water to provide additional assurance of maintaining primary containment integrity and function in the event of extended relief valve operation. The addition of Condition 2.c.(3) to DPR-62 requires Carolina Power & Light Company to undertake a program for seismic monitoring for a minimum of two years.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (The Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Proposed Issuance of Amendment to Facility Operating License in connection with the change in Technical Specifications was published in the FEDERAL REGISTER on June 19, 1975. We received no request for a hearing or petition for leave to intervene following this notice. In addition, "Order Modifying License and Revoking Order to Show Cause", adding Condition 2.c.(3) to DPR-62 was published in the FEDERAL REGISTER on July 16, 1975.

For further details with respect to this Amendment, see: (1) the application for amendment, dated April 3, 1975; (2) Amendment No. 3 to License No. DPR-62, with Change No. 3; (3) the Commission's related Staff Evaluation; and (4) "Order Modifying License and Revoking Order to Show Cause", dated July 9, 1975. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Southport - Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

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

Dated at Bethesda, Maryland, this 11th day of August, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor LIcensing

The enclosed pages 3.7-1, 3.7-2, 3.7-23, 3.7-23a, 3.7-23b and 3.7-24 are replacement pages to the Brunswick Steam Electric Plant, Unit 2, Operating License DPR-62, Appendix A for Change No. 3.

LIMITING CONDITIONS FOR OPERATION

3.7 Containment Systems

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. a. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.1.b.

   1) Minimum water volume - 87,600 ft^3
   2) Maximum water volume - 89,600 ft^3*
   3) Maximum suppression pool temperature during normal power operation - 95F
   4) Maximum suppression pool temperature during testing which adds heat to the suppression pool - 105°F
   5) Maximum suppression pool temperature during reactor power operation, defined as anytime the reactor is critical and above 1% of the licensed power level - 110°F (Initiate a scram if 110°F is reached)
   6) Maximum suppression pool temperature following a scram from continuous power operation without initiating plant depressurization-120°F (Depressurize reactor vessel at normal cool down rate to <200 psig if 120°F temperature is reached)

SURVEILLANCE REQUIREMENTS

4.7 Containment Systems

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication that a significant amount of heat is being added to the pressure suppression pool, the pool temperature shall be continuously monitored, observation of the pool temperature shall be maintained and the temperature logged every five minutes until the heat addition is terminated.

   c. Whenever there is indication that relief valve operation occurred with the pressure suppression pool temperature in excess of 160°F and the nuclear system pressure in excess of 200 psig a visual examination of selected ECCS suction line penetrations of the suppression pool enclosure shall be conducted before resuming power operation.

Change No. 3

Change No. 3

* Does not apply when the reactor is at atmospheric pressure and vented.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A.1.a Primary Containment (Cont'd)	4.7.A.1 Primary Containment (Cont'd)
7) In order to continue reactor power operation after being on RCIC, HPCI, or relief valve operation, the suppression chamber temperature must be reduced to 95F within 24 hours following the return to reactor power operation.	
b. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212 F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed five Mwt.	
2. Containment Leak Rate Testing	2. Containment Leak Rate Testing
a. Preoperational - General	a. Preoperational - General
The preoperational measured leakage rate L_{tm} shall not exceed 75 percent of the allowable test leakage rate L_t if: $L_{tm}/L_{am} \leq 0.7$ then: $L_t = L_a(L_{tm}/L_{am})$ L_a = design basis accident leakage rate which shall not exceed 0.5 percent by weight of the volume of the containment atmosphere at 49 psig per 24 hours. if: $L_{tm}/L_{am} > 0.7$ then: $L_t = L_a\ (P_t/P_a)^{1/2}$	The primary containment integrity shall be demonstrated by performing an integrated primary containment leak test (IPCLT) in accordance with the reduced pressure test program of Appendix J of 10CFR50, prior to initial unit operation at the test pressure of 49 psig (P_a), and 25 psig (P_t) to obtain the measured leak rates, L_{am} and L_{tm}, respectively. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves. The test duration shall not be less than 24 hours.

Change No. 3

August 1975

BASES:

3.7.A & 4.7.A Primary Containment (Cont'd)

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 49 psig which is below the design pressure of 62 psig. Maximum water volume of 89,600 ft^3 results in a downcomer submergence of 4'4" and the minimum volume of 87,600 ft^3 results in a submergence approximately four inches less. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170° F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in Specification 3.5.F.

Under full power operation conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 135°F immediately following blowdown which is below the temperature 170°F used for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings. Change No. 3

BASES:

3.7.A & 4.7.A Primary Containment (Cont'd)

Because of the large volume and thermal capacity of the pressure suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the pressure suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Change No. 3

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

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Change No. 3

BASES:

3.7.A & 4.7.A Primary Continment (Cont'd)

If a loss of coolant accident were to occur when the reactor water temperature is below approximately 330 F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212 F, shall be governed by this Specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212 F provides additional margin above that available at 330 F.

Inerting

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

Vacuum Relief

The purpose of the vacuum relief valves is to protect the primary containment vessel from external overpressure. The vacuum relief system from the pressure suppression chamber to Reactor Building consists of two 100 percent vacuum breakers. Operation of either valve will provide sufficient air flow from the Reactor Building to the suppression chamber to prevent the containment vessel from reaching to design external pressure rating of two psig. One