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FROM: Carolina Power & Light Co. Raleigh, N.C. E.E. Utley	DATE OF DOC 4-3-75	DATE REC'D 4-5-75	LTR xxx	TWX	RPT	OTHER
TO: Mr. Walter R. Butler	ORIG 3-signed	CC	OTHER	SENT AEC PDR xxxx SENT LOCAL PDR xxx		
CLASS	UNCLASS xxxx	PROP INFO	INPUT ✓	NO CYS REC'D 40	DOCKET NO: 50-261	

DESCRIPTION:

Ltr notarized 4-3-75 trans the following:

ENCLOSURES:

Amdt/OL change to tec-specs consisting of limitations and surveillance requirements to mitigate the potential for damage to the, torus, or suppression pool, from the effects of the steam quenching vibration phenomena
....

PLANT NAME: H.B. Robinson #2

FOR ACTION/INFORMATION 4-9-75 JGB

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CP&L

Carolina Power & Light Company

April 3, 1975

File: NG-3514 (B)

Serial: NG-75-436

Mr. Walter R. Butler, Chief
Light Water Reactors Branch 1-2
Division of Reactor Licensing
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

APR 1975
U.S.
Mail Section

Dear Mr. Butler:

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
LICENSE NO. DPR-62
REQUEST FOR LICENSE AMENDMENT - REVISIONS TO TECHNICAL SPECIFICATIONS

In accordance with the Code of Federal Regulations, Title 10, Parts 50.59 and 50.90, Carolina Power & Light Company submits a proposed revision to the Technical Specifications for its Brunswick Steam Electric Plant, Unit No. 2. The revision, which is attached to this letter, provides temperature limitations and surveillance requirements to mitigate the potential for damage to the torus, or suppression pool, from the effects of the steam quenching vibration phenomena.

This revision is filed in response to your request of February 18, 1975 and is part of the necessary information to be filed in order to assure the integrity of the suppression pool considering the effects of both the steam vent cleaning and steam quenching vibration phenomena. The additional information which you have requested will be filed on a schedule compatible with the schedule attached to your request.

The temperature limits and surveillance requirements in the attachment are based on the recommendations forwarded to you by the letter of Reference 1. A modification to the surveillance requirement concerning external visual examination of the torus (pressure suppression chamber) is necessary in the case of the Brunswick Plant due to the inaccessibility of the external surface resulting from its being almost wholly encased in a concrete support structure. An alternate means of examination which will ensure continued safe plant operation is thus proposed. To provide additional assurance of pool integrity, plant Emergency Instruction EI-40 has been established which provides limits on suppression pool conditions. The contents of EI-40 have been covered in detail in our December 10, 1974 reply to RO Bulletin 74-14. Additional procedure changes based on the limits in the attachment to this letter will be implemented prior to the Technical Specifications being issued on August 29, 1975. We trust the above information is adequate for your review of our submittal.

As required by Commission regulations, this submittal is signed under oath by a duly authorized officer of the Company.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:bn

cc: Mr. N. B. Bessac
Mr. P. W. Howe
Mr. E. G. Hollowell
Mr. R. E. Jones
Mr. D. B. Waters

Sworn to and subscribed before me this 3rd day of April

Nancy A. Stephens (Yancey)
Notary Public

My commission expires: June 29, 1976

NANCY A. STEPHENS NOTARY PUBLIC WAKE COUNTY, N.C.

Reference 1: Letter, E. G. Case, USAEC, from I. F. Stuart, GE Company, dated December 20, 1974.

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.

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 observed 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.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 Containment Systems	4.7 Containment Systems
Applicability:	Applicability:
Applies to the operating status of the primary and secondary containment systems.	Applies to the primary and secondary containment integrity.
Objective:	Objective:
To assure the integrity of the primary and secondary containment systems.	To verify the integrity of the primary and secondary containment.
Specification:	Specification:
A. Primary Containment	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. a. The suppression chamber water level and temperature shall be checked once per day.
1) Minimum water volume - 87,600 ft^3	b. Whenever there is indication that a significant amount of heat is being added to the pressure suppression pool, observation of the pool temperature shall be maintained and the temperature logged every five minutes until the heat addition is terminated.
2) Maximum water volume - 89,600 ft^3*	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 psia, a visual examination of selected ECCS suction line penetrations of the suppression pool enclosure shall be conducted before resuming power operation.
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	

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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 6) Maximum suppression pool temperature following a scram from continuous power operation without initiating plant depressurization - 120°F	

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 48 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	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.
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.	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.
if: $L_{tm}/L_{am} > 0.7$ then: $L_t = L_a\ (P_t/P_a)^{1/2}$	