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DESCRIPTION:	ENCLOSURES:
Ltr trans the following.....	Request for Lic Amdt as Tech Spec Rev re removal of limitations on steam generator water chemistry presently contained in Tech Specs......notarized 9-6-74....

PLANT NAME: HBROBINSON

DO NOT REMOVE

ACKNOWLEDGED

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CP&L

Carolina Power & Light Company

September 4, 1974

50-261

File: NG-3514 (R)

Serial: NG-74-1020

DOCKETED USAEC SEP 9 1974 REGULATORY MAIL SECTION DOCKET CLERK

RECEIVED SEP 9 1974

Mr. Edson G. Case, Deputy Director
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Case:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
REQUEST FOR LICENSE AMENDMENT
REVISION OF TECHNICAL SPECIFICATIONS

In accordance with the Code of Federal Regulations, Title 10, Part 50.59, Carolina Power & Light Company requests a revision to the Technical Specifications for its H. B. Robinson Unit No. 2 Plant. The revision concerns the removal of limitations on steam generator water chemistry presently contained in the Technical Specifications.

During the latter part of 1972 and the first part of 1973, discussions were held between the AEC and Carolina Power & Light Company concerning the consequences of steam generator tube degradation that had been experienced in the Robinson Plant in May of 1972. The major area of concern was the increased site boundary dose resulting from the failure of collapsed fuel clad sections and subsequent failure of degraded steam generator tubes during the power and pressure transients created as a result of a steam pipe rupture. These discussions resulted initially in a requirement to perform monthly tests of the steam generator tube integrity, with a primary-secondary differential pressure greater than that experienced during a steam break accident. After successfully completing three of these tests, Carolina Power & Light Company later conducted a complete eddy current investigation of the steam generator tubing during the first refueling outage of Robinson in March-May, 1973, resulting in two additional tubes being plugged but not further signs of degradation. As a result of this investigation, which was reported to the AEC on May 11, 1973, along with a copy of the then current Westinghouse steam side water chemistry control specifications as additional information, revised Technical Specifications were drafted by the AEC and issued to Carolina Power & Light Company. These specifications addressed pressure testing of the steam generator (Specification 4.7.2) and specifications for phosphate chemistry and oxygen concentration (Specifications 3.4.1.f, g, h, i and revised Specification 3.4.3). It should be noted that at the time these specifications were issued, the Robinson core still contained fuel with the potential for collapsed clad sections, as opposed to the fuel presently contained in the core which will not experience collapse.

9302

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Eddy current investigations were again conducted in essentially all steam generator tubes during the second refueling outage in May-June, 1974. The results, which were reported to the AEC on July 10, 1974, showed that additional tube degradation had occurred, of a different mechanism than was experienced in 1972 but still related to phosphate chemistry control on the secondary side.

Even though it has been shown that water chemistry control is of prime importance in the mechanisms for steam generator tube degradation, Carolina Power & Light Company feels it is inappropriate to include such control parameters as Technical Specifications, since these parameters will most likely be reevaluated and refined as more operating data is gathered. For example, the present Technical Specifications require that the minimum concentration of 10 ppm phosphate be maintained in the steam generators above $350^{\circ}F$. During operation up until December, 1973, Westinghouse recommended that the Na/PO_4 ratio be maintained between 2.0 to 2.6. After an outage in November, 1973, the recommendation was modified to maintain Na/PO_4 ratio of 2.3 to 2.6. The limitations of 10 ppm minimum phosphate concentration interfered with the steam generator chemistry changeover procedure during the plant startup in December, 1973, which would provide a means of removing the low ratio phosphate species believed to exist in the sludge layer present on the steam generator tube sheet. Removal of the low ratio phosphate, which would have been achieved by blowdown and securing phosphate feed at the hot shutdown condition, had to be terminated prematurely to avoid violation of the Technical Specification. During plant startup after the 1974 refueling outage, a variance of this specification had to be obtained to allow proper establishment of water chemistry.

Carolina Power & Light Company, in close association with Westinghouse, is continually attempting to avert steam generator tube degradation by following recommended steam generator chemistry control specifications. However, applying these as Technical Specifications before the steam generator chemistry is better understood and established may be at variance with the objective of protecting steam generator tube integrity through proper chemistry control. An additional example to the one cited above is the recent recommendation by Westinghouse to their utility customers to change from phosphate water chemistry control to an all-volatile treatment. This changeover will be recommended for operating plants as soon as suitable procedures can be worked out. Carolina Power & Light Company intends to consider this recommendation, which would require that Technical Specifications peculiar to phosphate chemistry control be removed.

We are in receipt of the AEC letter of July 18, 1974, requiring that Technical Specifications be established which provide for a program of steam generator tube inspection and the reporting of results, in compliance with Regulatory Guide 1.83. We feel that these Technical Specifications, when they are developed, will provide the assurance that the health and safety of the public will not be unduly jeopardized by the effects of steam generator tube degradation better than specifications on water chemistry. We, therefore, request that specifications on steam generator water chemistry be removed

from the H. B. Robinson Unit No. 2 Technical Specifications, as presented in the attached page changes to the Specifications.

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

Sworn to and subscribed before me this 6th day of September, 1974.

Nancy A. Stephens (Yancey)

My commission expires: June 29, 1976

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

DBW:mvp
Attachment

cc: Messrs. N. B. Bessac
T. E. Bowman
W. B. Howell
J. B. McGirt
D. V. Menscer
D. B. Waters

3.4 SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

3.4.1 The reactor coolant shall not be heated above 350°F unless the following conditions are met:

(a) A minimum turbine cycle steam relieving capability of twelve (12) main steam safety valves operable.

(b) Two of the three auxiliary feedwater pumps must be operable.

(c) A minimum of 35,000 gallons of water in the condensate storage tank and an unlimited water supply from the lake via either leg of the plant Service Water System.

(d) Essential features including system piping and valves directly associated with the above components are operable.

(e) The main steam stop valves are operable and capable of closing in five seconds or less.

3.4.2 The iodine-131 activity on the secondary side of a steam generator shall not exceed 0.24 uCi/cc.

3.4.3 If, during power operations, any of the specifications in 3.4.1 or 3.4.2 above cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The twelve main steam safety valves have a total combined rated capability of 10,068,845 lbs/hr. The total full power steam flow is 9,589,000 lbs/hr., therefore, twelve (12) main steam safety valves will be able to relieve the total steam flow if necessary.[1] Following a loss of load, which represents the worst transient, steam flows are below the total capacity of the 12 safety valves. Therefore, overpressurization of the secondary system is not possible.

In the unlikely event of complete loss of turbine-generator and offsite electrical power to the plant, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps operated from the diesel generators and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant.[2] The minimum amount of water in the condensate storage tank is the amount needed for at least 2-hours operation at hot standby conditions. If the outage is more than 2 hours, deep well or Lake Robinson water may be used.

An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

The limit on secondary coolant iodine-131 specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 is the dominant isotope because of its low MPC in air and because the other shorter lived iodine isotopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation thyroid dose at the site boundary is then:

$$\text{Dose (Rem)} = C \cdot V \cdot B \cdot DCF \cdot X/Q \cdot 0.1$$

Where: C = secondary coolant I-131 specific activity

= 0.24 curies/m^3 (uc/cc)

V = equivalent secondary coolant volume released = 135 m^3

B = breathing rate = 3.47×10^{-4} m^3/sec.

X/Q = atmospheric dispersion parameter = 8.9×10^{-4} m^3/sec.

0.1 = equivalent fraction of activity released

The resultant thyroid dose is less than 1.5 Rem.

References

(1) FSAR - Section 10.3

(2) FSAR - Section 14.2.5

Figure 3.4-1 Deleted

4.7 SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to periodic testing of secondary system components and surveillance of secondary coolant.

Objective

To verify the ability of secondary system components to function as required and to prevent system degradation.

Specification

4.7.1 The main steam stop valves shall be tested at each refueling interval of each 15 ± 3 months, whichever occurs first. Closure time of five seconds or less shall be verified. The valves are tested under no flow and at no load conditions.

4.7.2 The steam generators shall be pressure tested at each refueling interval or each 15 ± 3 months, whichever occurs first. Either of the following test conditions may be applied:

(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.

Basis

The main steam stop valves serve to limit an excessive Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with expected response time for instrumentation as detailed in the steam line break incident analysis.

References

FSAR - Section 10.4

FSAR - Section 14.2.5