MAR 28 1984

Docket No. 50-261

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See correction letter of 4/20/84

Mr. E. E. Utley, Executive Vice President
Power Supply and Engineering & Construction
Carolina Power and Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

The Commission has issued the enclosed Amendment No. 78 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated October 14, 1983.

The amendment revises the Technical Specification as follows:

1. One change incorporates Section 4.05 of the Westinghouse Standard Technical Specification requirements regarding testing requirements of Section XI of the ASME code and deleting the detailed requirements covered by Section XI.

2. One change revises nomenclature to be consistent with HBR-2 FSAR and plant conditions with regard to turbine trip setpoints.

3. One change adds limitations not currently included in the Technical Specification but included in Section 7.2.1.1.1 of the FSAR with regard to Steam Flow/Feedwater Flow Mismatch.

4. One change reinstates the frequency for testing prior to startup which was contained in the Technical Specification prior to Amendment 65.

5. One change revises Technical Specification Table 4.1-3 to achieve consistency within the specification.

We have made minor change Technical Specification 6.5.1.6.3. These changes have been discussed with and agreed to by members of your staff.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY

Glode Requa, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 78 to DPR-23
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL
CParrish
3/27/84

ORB#1:DL
GRequa:ps
3/27/84

C-ORB#1:DL
SVarga
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OELD
C Bachmann
3/27/84

AD:OR:DL
GLainas
3/ /84

Mr. E. E. Utley
Carolina Power and Light Company

H. B. Robinson Steam Electric
Plant 2

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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. DPR-23

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

   A. The application for amendment by Carolina Power and Light Company (the licensee) dated October 14, 1983, 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 I;

   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;

   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; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendx A, as revised through Amendment No. 78 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 28, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 78 FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.8-3	3.8-3
3.13-1	3.13-1
------------	3.13-5
4.4-2	4.4-2
4.4-8	4.4-8
4.13-1	4.13-1
4.13-2	4.13-2
4.13-3	4.13-3
4.13-4	4.13-4
----------	4.13-5
4.15-2	4.15-2
6.2-1	6.2-1
------------	6.2-1a
6.5-6	6.5-6
6.5-11	6.5-11
6.13-1	6.13-1

j. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease; work shall be initiated to correct the conditions so that the specified limits are met; and no operations which may increase the reactivity of the core shall be made.

k. The reactor shall be subcritical as required by 3.10.8.3.

3.8.2 The Spent Fuel Building Filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal absorber banks shall show $\geq$ 99 percent DOP removal and $\geq$ 99 percent halogenated hydrocarbon removal.

b. Verification by way of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show $\geq$ 90 percent radioactive methyl iodide removal in accordance with test 5.b of Tale 5-1 of ANSI/ASME N509-1976 except that $\geq$ 70 percent relative humidity air is required.

c. All filter system fans shall be shown to operate within ± 10% of the design flow.

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be $\leq$ 70 percent.

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

3.13 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to shock suppressors (snubbers) as shown in Tables 3.13-1 and 3.13-2.

Objectives

To provide for limiting conditions for operation which ensure the operability of snubbers during plant operation, such that normal operation or plant transients requiring operation of the snubbers will not result in consequences more severe than those previously analyzed.

Specification

3.13.1 During all modes of operation except cold shutdown and refueling, all snubbers specified in Tables 3.13-1 and 3.13-2 shall be capable of performing their intended function in the required manner (operable) except as described below:

a. When the reactor is at hot shutdown or at power and a snubber is determined to be inoperable, an engineering analysis must be conducted within 72 hours to determine if the snubber's inoperability has adversely affected the supported component. If so, the supported component shall be declared inoperable and appropriate action shall be taken in accordance with the appropriate Technical Specification. If the supported component has not been adversely affected, (1) an analysis shall be performed to determine if the supported component could be damaged during a future event and, if so, the snubber shall be repaired or replaced within 72 hours of finding it inoperable, or (2) the supported component shall be declared inoperable until the snubber is repaired or replaced and appropriate action shall be taken in accordance with the appropriate Technical Specification. If the analysis demonstrates that the snubber is not needed for the supported component to be adequately protected during normal operation and design events, reactor operation shall continue and the snubber shall be repaired on a routine basis.

b. If a snubber is determined to be inoperable while the reactor is in cold shutdown, the snubber (if needed for supported component protection) shall be repaired and reinstalled or replaced prior to reactor startup.

TABLE 3.13-2
SAFETY RELATED MECHANICAL SNUBBERS

Snubber No.	Location	Elevation	Snubbers in High Radiation Area During Shutdown*	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
31	Steam Supply to Aux. Feed Pump (Point #4)	265'				X
32	Steam Supply to Aux. Feed Pump (Point #24)	265'				X

d. The test shall be performed without preliminary leak detection survey or leak repairs. Leak repairs, if required to meet the acceptance criteria during the integrated leakage test, shall be preceded by local leakage rate measurements. The leakage rate difference, prior to and after repair and corrected to the test pressure (P_t) shall be added to the final integrated leakage rate result.

e. All mechanical fluid systems which, under post-accident conditions, become an extension of the containment pressure boundary shall be vented to the containment atmosphere prior to the test. Closure of containment isolation valves shall be accomplished by the normal mode of operation.

f. Acceptance Criteria

(1) The maximum allowable leak rate L_p shall not exceed 0.1 weight percent of the contained air per 24 hours at the test pressure of 42 psig (P_p).

(2) The allowable test leak rate at a test pressure of 21 psig, $L_t(21)$ shall not exceed the value established as follows:

$$L_t(21) = 0.1\ L_m(21)/L_m(42)$$

or

$$= 0.1\ (P_t/P_p)^{1/2}$$

c. Notification of the pending test, either of a sample tendon or the containment structural test, along with detailed acceptance criteria shall be forwarded to the Nuclear Regulatory Commission two months prior to the actual test. Within six months of conducting the test, a report and evaluation shall be submitted to the NRC.

Basis

The containment is designed for an accident pressure of 42 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 42 psig is 263°F.

Prior to initial operation, the containment was strength tested at 48.3 psig and then was leak-tested. The acceptance criterion for this preoperational leakage rate test was established as 0.08 weight percent of the contained air per 24 hours at 42 psig. This acceptable leakage rate was equivalent to a 0.1 weight percent of the contained steam-air atmosphere per 24 hours at 42 psig and 263°F. The acceptance criteria for Integrated Leakage Rate Tests (ILRTs) is now established as 0.1 weight percent of the contained air per 24 hours at 42 psig. This value is reduced to 0.075 weight percent of the contained air per 24 hours per Section 4.4.1.1.f.(3) to provide added conservatism to the test results. The leakage rate at 42 psig must not exceed this reduced value. These leakage rates are consistent with the construction of the containment,[2] which is equipped with a penetration pressurization system which pressurizes penetrations, double gasketed seals, and some isolation valve spaces. The channels over all of the containment liner welds were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10% per 24 hours at 42 psig and 263°F. With this leakage rate and with minimum containment engineered safety features operating, the public exposure would not exceed 10 CFR 100 guideline values in the event of the design basis accident.[3]

4.13 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to shock suppressors (snubbers) listed in Tables 3.13-1 and 3.13-2.

Objectives

To ensure the continued operability of snubbers by periodic surveillance.

Specification

4.13.1 Visual Inspection

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment and all mechanical snubbers shall be visually inspected in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
≥8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

b. All hydraulic snubbers whose seal materials are other than ethylene propylene, Viton "A", or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

c. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.13.1.a, it shall be assumed that the facility had been on a 6 month inspection interval.

d. Visual inpsection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure,and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.13.2; (2) the cause of the rejection is clearly established and remedied for that particular snubber. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

4.13.2 FUNCTIONAL TESTING

a. Once each refueling cycle, a representative sample of approximately 10 percent of the hydraulic snubbers shall be functionally tested for operability including verification of proper piston movement, lock up and bleed rates. For each snubber found inoperable, an additional ten percent of the snubbers of that type shall be functionally tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, at least one mechanical snubber shall be functionally tested for operability including verification of proper piston movement, drag force, release rate, and actuating acceleration.

c. A representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25 percent of the snubbers in the representative sample shall include snubbers from the following categories:

   a. Snubbers within 5 feet of heavy equipment (valve, pump, steam generator, etc.).

   b. Snubbers within 10 feet of the discharge from a safety/relief valve.

d. The steam generator snubbers (500,000 lbs. ft. rated capacity) need not be removed for functional testing unless the visual inspection dictates that a snubber be removed for corrective maintenance. The testing requirement for these snubbers can be satisfied by testing the control unit (valve block) instead of the entire snubber.

e. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

f. If any snubber selected for functional testing either fails to lockup or fails to move; i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

g. For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order toensure that the supported component remains capable of meeting the designed service.

4.13.3 Snubber Service Life Monitoring

A record of the service life of each snubber listed on Tables 3.13-1 and 3.13-2, the date at which the designated service life commences and the installation and maintenance records on which the service life is based shall be maintained.

Once each refueling cycle, these records shall be reviewed to ensure that the service life will not be exceeded prior to the next review. If the service life of a snubber will be exceeded prior to the next scheduled review, the snubber's service life can be reevaluated in order to possibly extend it or the snubber shall be reconditioned or replaced. This reevaluation, replacement, or reconditioning shall be indicated in the records.

Basis

All safety-related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level (as applicable), and proper attachment of snubber to piping and structures.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Viton "A" and ethylene propylene seal material have been demonstrated by lab tests and operating experience to be compatible with nuclear plant operating environments.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment shall be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required visual inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

A snubber which appears inoperable as a result of a visual inspection may be declared operable if it passes a functional test and the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. For hydraulic snubbers these tests will include stroking of the snubbers to verify proper piston movement, lock up, and bleed rates. For mechanical snubbers these tests will include stroking of the snubbers to verify proper piston movement, drag force, release rate, and actuating acceleration. Ten percent of the snubbers listed on Tables 3.13-1 and 3.13-2 represent an adequate sample for such tests. Observed failures of these samples shall require testing of additional units.

Periodic functional testing of the stream generator snubbers (as a unit) is not required due to their large size and difficulty of removal. By testing the smaller and more easily removable control unit for each snubber, the operability of these large bore snubbers can be ensured.

When a snubber is found inoperable (visual or functional), an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber's mode of failure has imparted a significant effect or degradation on the supported component or system.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirements to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future adjustments of snubbers' service lives. The review of the snubber's service lives and necessary reconditioning or replacement shall take place once per operating cycle probably during the refueling outage.

d. Verify by way of a laboratory test that the system's carbon demonstrates a methyl iodide removal efficiency of $\geq$ 90 percent. The test shall be conducted in accordance with ANSI N509-1976, Table 5-1, Test 5b. The required carbon samples may be obtained by the following methods.

1. One sample obtained from a test canister designed to ANSI N509-1976. The sample must be at least two inches in diameter and with a length equal to or greater than the thickness of the cell's absorber bed.

2. Two samples obtained by emptying an adsorber cell and mixing the carbon thoroughly. The samples must be at least two inches in diameter and with a length equal to or greater than the thickness of the cell's adsorber bed.

4.15.2 At least once per operating cycle, the following test shall be performed:

a. Verify that the pressure drop across the combined HEPA filters and charcoal adsorber bank is <6 inches Water Gauge at system design flow rate ± 10 percent.

b. Verify that on a containment isolation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

4.15.3 After each complete or partial replacement of the carbon adsorber bank, perform the tests under Specification 4.15.1b.

4.15.4 After each complete or partial replacement of the HEPA filter bank, perform the tests under Specification 4.15.1c.

4.15.5 The associated fan unit in the Control Room filter system shall be verified operable monthly.

6.2 ORGANIZATIO"

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2-2 and:

a. The shift complement during hot operations shall consist of at least one Shift Foreman holding a Senior Reactor Operator's License, one Senior Control Operator holding a Senior Reactor Operator's License, two control operators each holding a Reactor Operator's license, two additional shift members, and one Shift Technical Advisor.

b. The shift complement during cold shutdown shall consist of at least one Shift Foreman holding a Senior Reactor Operator's License, one Control Operator holding a Reactor Operator's License and one additional shift member.

c. At least one licensed Operator shall be in the control room when fuel is in the reactor.

d. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown, and during recovery from reactor trips.

e. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

f. ALL CO ALTERATIONS after the initial f i loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

g. A Plant Fire Brigade of at least 5 members shall be maintained on site at all times. This excludes three members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency.

Science in engineering or related field or equivalent and two (2) years related experience.

6.5.1.6 Plant Nuclear Safety Committee (PNSC)

6.5.1.6.1 a. As an effective means for the regular overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) is established.

b. The committee shall function through the utilization of subcommittes, audits, investigations, reports, and/or performance of reviews as a group.

6.5.1.6.2 The PNSC shall be composed of the following:

Chairman - General Manager or designated alternate

Member - Manager - Operations and Maintenance or designated alternate

Member - Manager - Technical Support or designated alternate

Member - Assistant to General Manger

Member - Manager - Environmental & Radiation Control or designated alternate

Member - Director - QA/QC or designated alternate

6.5.1.6.3 Alternates shall be appointed in writing by the General Manager to serve on a temporary basis. All alternates shall, as a minimum, meet qualification criteria specified in Section 4.4 of ANSI N 18.1-1971 for professional-technical personnel or, for those disciplines not listed in Section 4.4, the equivalent of the Section 4.4 requirement.

6.5.1.6.4 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman or his designated alternate.

6.5.2.3 Personnel

a. Personnel assigned responsibility for independent reviews shall be specified in technical disciplines and shall collectively have the experience and competence required to review problems in the following areas:

(1) Nuclear power plant operations
(2) Nuclear engineering
(3) Chemistry and radiochemistry
(4) Metallurgy
(5) Instrumentation and control
(6) Radiological safety
(7) Mechanical and electrical engineering
(8) Administration controls
(9) Seismic and environmental
(10) Quality assurance practices
(11) Nondestructive Testing

b. The following minimum experience requirements shall be established for those persons involved in the independent safety review program:

(1) Manager of CNSS - Bachelor of Science in engineering or related field and ten (10) years' related experience, including five (5) years' involvement with operation and/or design of nuclear power plants.

(2) Reviewers - Bachelor of Science in engineering or related field or equivalent and five (5) years' related experience.

c. An individual may possess competence in more than one specialty area. If sufficient expertise is not available

6.13 HIGH RADIATI AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mr/hr but less than 1000 mr/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation exposure rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mr/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Foreman on duty, and/or the Radiation Control Foreman.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

Introduction

By letter dated October 14, 1984, Carolina Power and Light Company (the licensee) proposed revisions to the Technical Specifications of the H. B. Robinson Steam Electric Plant Unit No. 2. The proposed revisions would clarify the qualification requirements for alternate members of the Plant Nuclear Safety Committee (PNSC); make the list of technical disciplines for the Corporate Nuclear Safety Section (CNSS) Independent Review consistent with ANSI N18.7-1976; incorporate mechanical snubbers installed during the last refueling outage; incorporate Standard Technical Specifications (STS) terminology regarding the Administrative Control of High Radiation Areas; revised the Acceptance Criteria for the Integrated Leak Rate Test (ILRT); revise staffing specifications to be in accordance with requirements; and correct typographical errors and inconsistencies. The proposed revisions are discussed individually below.

ALTERNATES FOR PLANT NUCLEAR SAFETY COMMITTEE
(Specification 6.5.1.6.3)

Discussion and Evaluation

In their letter of October 14, 1983, CP&L proposed changes to the Administrative Controls Technical Specification 6.5.1.6.3, noting that the qualification requirements specified for alternate plant nuclear safety committee (PNSC) members did not represent all of the functional areas which compose the PNSC. The qualification requirements specified that alternates shall, as a minimum, meet the qualifications specified for professional-technical personnel in Section 4.4 of ANSI-N18.1-1971 The licensee's proposed change which stated: ". . . All alternates shall, as a minimum, meet equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI-N18.1-1971", was not sufficiently clear to determine how alternates not listed in Section 4.4 should be qualified. Therefore, clarifications were discussed with and agreed to by the licensee. The clarified wording is: All alternates shall, as a minimum, meet qualification criteria specified in Section 4.4 of ANSI-N18.1-1971 for professional-technical personnel or, for those disciplines not listed in Section 4.4, the equivalent of the Section 4.4, requirement.

This clarifies the qualification requirements for alternate PNSC members and will not result in a change to facility operations. This change is administrative and therefore does not involve a significant hazards consideration.

Conclusion

With the clarifications agreed to by the licensee, the submitted administrative change to the specification is accepted and amends the existing specification.

CORPORATE NUCLEAR SAFETY SECTION INDEPENDENT REVIEW CRITERIA
(Specification 6.5.2.3)

Discussion and Evaluation

In their letter of October 14, 1983, CP&L proposed a change to the Administrative Controls Technical Specification 6.5.2.3 which would add the technical area of Nondestructive Testing to the list of areas in which members of the Corporate Nuclear Safety Section are required to collectively possess the experience and competence necessary to perform reviews.

The proposed change is administrative in nature and would allow the licensee to conform to the guidance of ANSI N18.7-1976 with respect to the inclusion of Nondestructive Testing in the list. The change constitutes an additional requirement for the independent review group as listed in the Technical Specifications and does not involve a significant hazards consideration.

Conclusion

The proposed change is accepted as submitted and amends the corresponding Specification.

SNUBBERS
(Specifications 3.13 and 4.13)

Discussion and Evaluation

In their letter of October 14, 1983, CP&L proposed changes to Technical Specifications which were necessitated by the addition of two safety-related mechanical snubbers to the auxiliary feedwater system. These are the first safety-related mechanical snubbers to be installed at Robinson. The changes to the Technical Specifications identify surveillance requirements for the added safety-related shock suppressors. The proposed changes were based on guidance provided by the NRC and do not involve a significant safety hazards consideration.

Conclusion

The proposed changes are accepted as submitted and are incorporated into the Technical Specifications.

HIGH RADIATION AREA KEY CONTROL
(Specification 6.13)

Discussion and Evaluation

In their letter of October 14, 1983, CP&L proposed an administrative change to Technical Specification 6.13 which would allow for control of High Radiation Area Keys by the Radiation Control Foreman. This proposed change conforms to the guidance used in Standard Technical Specifications by allowing these keys to be administratively controlled by the Shift Foreman on duty and/or the Radiation Control Foreman.

The change revises the administrative controls providing consistency with Standard Technical Specifications, enhancing control of access to High Radiation Areas, and allowing a reduction in the administrative burden on the Shift Foreman. The proposed change does not involve a significant safety hazards consideration.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications.

ACCEPTANCE CRITERIA FOR INTEGRATED LEAK RATE TEST
(Specification 4.4.1.1.f)

Discussion and Evaluation

In their letter of October 14, 1983. CP&L proposed changes to Technical Specification 4.4.1.1.f which would increase the maximum allowable leak rate for the containment vessel during testing from 0.08 percent per day to 0.1 percent per day. The 0.08 value represents the leakage criteria at a containment internal environmental temperature of 120°F, the expected air temperature during reactor operation. The 0.1 value is the equivalent leakage rate at a containment internal environmental temperature of 263°F, the expected temperature of the steam-air mixture at the peak accident pressure. The 0.08 value was used in accordance with the previously existing requirements of the AEC Technical Safety Guide (Revised Draft - December 15, 1966) in order to correct test temperature during the Integrated Leak Rate Test to accident temperature. However, issuance of 10 CFR 50, Appendix J superseded the requirements of the AEC Technical Safety Guide. Appendix J does not require the 20 percent reduction in leak rate from 0.1 to 0.08, but rather requires the measured leakage to be less than 75 percent of the maximum allowable leak rate. This 25 percent reduction is already included in the Robinson Technical Specifications. Therefore, to eliminate the redundant reduction for the maximum allowable leak rate for the containment vessel, and to comply with the requirement of NRC Standard Review Plan 6.2.6 which specifies a minimum acceptable design containment leakage rate of not less than 0.1 percent per day, the licensee has proposed changing the Technical Specification 4.4.1.1.f leakage value from 0.08 percent per day to 0.1 percent per day.

The proposed change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequences of an accident previously evaluated, or create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in the margin of safety. This change does not involve a significant hazards consideration.

Conclusion

The proposed change is accepted as submitted and is incorporated into the Technical Specifications.

METHYL IODIDE
(Specifications 3.8.2.b and 4.15.1.d)

Discussion and Evaluation

CP&L proposed, in their letter of October 14, 1983, administrative changes to Technical Specifications 3.8.2.b and 4.15.1.d. In each case the Specifications reference a laboratory test for "methyl iodine". The correct term for the type of laboratory testing actually required and performed is "methyl iodide". These proposed changes correct a typographical error and are purely administrative in nature. They do not involve a significant hazards consideration.

Conclusion

The proposed changes are accepted as submitted and are incorporated into the Technical Specifications.

SHIFT STAFFING
(Section 6.2.2)

Discussion and Evaluation

In their letter of October 14, 1983, CP&L proposed changes to Technical Specification 6.2.2 regarding the composition and manning of the shift staff. The licensee's proposed changes would relax the required availability of the Shift Technical Advisor (STA), and add requirements for an additional shift member and an additional Senior Reactor Operator during hot operations.

The licensee currently requires that an STA be available for duty at all times. This requirement is more restrictive than NRC regulations and staff guidance. NUREG-0737, Clarification of TMI Action Plan, Item I.A.1.1, Shift Technical Advisor, requires that an STA be available for duty when the plant is operating in Modes 1-4. The proposed change in STA staffing would make the Technical Specification consistent with NUREG-0737. The proposed added shift manning requirements will make the Technical Specification consistent with 10 CFR 50.54(m)(2) and Section I.A.1.3 of NUREG-0737. These proposed additions constitute additional restrictions on the shift complement not presently in Technical Specifications and conform to recent revisions in the regulation as stated. These changes do not involve a significant hazard consideration.

Summary

The proposed changes are accepted as submitted and are incorporated into the Technical Specifications.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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: March 28, 1984

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