Attachment 1

ALTERNATES FOR PNSC

Details

Technical Specification 6.5.1.6.3 states, in part, that alternates shall, as a minimum, meet the qualifications specified for professional-technical personnel in Section 4.4 of ANSI-N18.1-1971. The groups defined in Section 4.4 do not represent all of the functional areas which compose the PNSC. In order to define the qualification requirements for groups of professional-technical personnel which are not represented in Section 4.4, we request that a revision be made to TS 6.5.1.6.3 to provide qualification requirements equivalent to those for the groups of professional-technical personnel which are represented in Section 4.4.

Summary of Safety Analysis and Significant Hazards Determination

This change clarifies the qualification requirements for alternate PNSC members and will not result in changes to facility operations. This change constitutes an additional limitation not presently included in the Technical Specifications and therefore does not involve a significant hazards consideration.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety.

Affected TS Page: 6.5-6

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 equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI N18.1-1971

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.

CNSS INDEPENDENT REVIEW CRITERIA

Details

The members of the CNSS, who perform independent reviews, are required to collectively possess the experience and competence necessary to review matters in the technical areas listed in TS 6.5.2.3.a. During a recent corporate audit it was noted that the existing list is not consistent with the list identified in ANSI N18.7-1976 in that Nondestructive Testing is not included. We request that this list be revised to incorporate Non-destructive Testing.

Summary of Safety Analysis and Significant Hazards Determination

This change will allow the licensee to conform to the guidance of ANSI N18.7-1976 with respect to non-destructive testing. This change constitutes an additional requirement for the independent review group as listed in the technical specifications and does not involve a significant hazards consideration.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety.

Affected TS Page: 6.5-11

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

Attachment 3

SNUBBERS

Details

Two mechanical shock suppressors (snubbers) were added to the Auxiliary Feedwater (AFW) System to provide additional support as a result of a modification which replaced the steam inlet valves to the steam driven auxiliary feedwater pump. These are the first safety-related mechanical snubbers to be installed at HBR2. Existing pages 3.13-1, 4.13-1, 4.13-2, 4.13-3, 4.13-4 and 4.13-5 were revised and a new page was created, incorporating the new Table 3.13-2, Safety Related Mechanical Snubbers. Mr. D. G. Eisenhut's letter dated November 20, 1980 concerning snubber surveillance was used as guidance in development of these revisions.

Summary of Safety Analysis and Significant Hazards Determination

This change identifies surveillance requirements for additional safety related shock suppressors recently added to the AFW system. These suppressors were added to provide additional support as a result of modifications to the AFW system and to enhance system safety. The snubbers are being included in the TS to ensure that appropriate test requirements are met. These changes to the TS were based on guidance provided by the NRC and provide surveillance requirements for equipment not previously incorporated in the Technical Specifications. This change does not involve a significant hazards consideration.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety.

Existing TS Affected Pages:	3.13-1 4.13-1 through 4.13-4
Revised Pages Attached:	3.13-1, 3.13-5 4.13-1 through 4.13-5

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

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.

Attachment 4

HIGH RADIATION AREA KEY CONTROL

Details

Section 6.13, High Radiation Area, Page 6.13-1, was revised to provide for additional administrative control of High Radiation Area keys. The STS state that the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the Plant Health Physicist. We have revised this section to include the Shift Foreman on duty and/or the Radiation Control Foreman. We believe that this will allow a reduction in the administrative burden on the Shift Foreman while providing adequate control at the appropriate level with the Radiation Control Foreman.

Summary of Safety Analysis and Significant Hazards Determination

This change incorporates additional administrative controls while providing consistency with standard TS therefore enhancing control of access to high radiation areas. Additional controls added to Technical Specifications do not involve a significant hazards consideration.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety.

Affected TS Page: 6.13-1

6.13 HIGH RADIATION 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.

ACCEPTANCE CRITERIA FOR ILRT

Details

Section 14.3.5 of the HBR2 Final Safety Analysis Report (FSAR) specifies a maximum allowable leak rate for the containment vessel of 0.08 weight percent per day. This reduction from 0.1 percent per day to 0.08 percent per day was taken in accordance with the then existing requirements of the AEC Technical Safety Guide (Revised Draft - December 15, 1966) in order to correct test temperature during the ILRT to accident temperature. However, issuance of 10 CFR 50, Appendix J supersedes the requirements of the AEC Technical Safety Guide and does not require a reduction in the maximum allowable leak rate for the containment vessel to account for accident temperature. In lieu of this reduction, Appendix J of 10 CFR 50 requires the measured leakage to be less than 75 percent of the maximum allowable leak rate. This requirement is already included in the HBR2 TS. 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 specify a minimum acceptable design containment leakage rate of not less than 0.1 percent per day, we have revised HBR2 TS 4.4.1.1.f (1) and (2) values from 0.08 percent per day to 0.1 percent per day. This was discussed with members of NRC staff on February 1, 1983.

Summary of Safety Analysis and Significant Hazards Analysis

The present TS allowed leakage criteria for the containment ILRT imposes a double penalty; specifically, a reduction of 20% (0.1% to 0.08%) is imposed to equate test conditions (air leakage) to a post-accident condition (steam-air mixture leakage). Also, another 25% reduction in acceptance criteria is imposed to allow for a possible increase in leakage rates between tests. This buffer is intended to ensure the acceptable leakage is not exceeded during operation. This TS revision will delete the 20% adjustment for test air versus post-accident steam-air mixture and make the TS consistent with 10 CFR 50 Appendix J. Members of NRR Containment Systems Branch stated their agreement with the plant position to elimiate the double conservatism penalty during a conference call.

An unreviewed safety question does not exist due to the following:

1) The 20% adjustment (0.10% to 0.08%) imposes an unnecessary penalty no longer required by the NRC;
2) The reduction of 25% for the possible leak rate increase between tests will remain in place;
3) The acceptance criteria change requested will not allow the FSAR DBA allowed leakage of 0.1% weight per day to be exceeded; and
4) This change makes the license conform to changes in the regulations which results in very minor changes to facility operations clearly in keeping with the regulations.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety. This change does not involve a significant hazards consideration.

Affected TS Pages: 4.4-2 and 4.4-8

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 the 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]

METHYL IODIDE

Details

Sections 3.8.2.b and 4.15.1.d presently discuss a laboratory test for methyl iodine. The correct term for the type of laboratory testing actually required, and the type of testing performed at HBR2, is methyl iodide instead of methyl iodine. Therefore, the typographical errors in the above sections are being changed to reflect the correct type of testing to be performed.

Summary of Safety Analysis and Significant Hazards Determination

This change corrects a typographical error and is purely administrative in nature and therefore does not involve a significant hazards consideration.

Based on the above, CP&L has determined that this change does not constitute an unreviewed safety question, nor does it involve a significant increase in the probability or consequence 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 margin of safety.

Affected TS Pages: 3.8-3
4.15-2

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 with $T_{avg} \leq 140°F$.

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.

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.

SHIFT STAFFING

Details

NUREG-0737, Clarification of TMI Action Plan, Item I.A.1.1, Shift Technical Advisor (STA), requires that an STA be available for duty when the plant is operating in Modes 1-4. However, Section 6.2.2.a of HBR2 TS requires one STA to be available for duty at all times. In addition, NUREG-0737 Item I.A.1.3, Shift Manning, requires that two auxiliary operators be on shift; HBR2 TS presently require only one "additional shift member" who must be at least an auxiliary operator. Therefore, Page 6.2-1 has now been revised to be in agreement with these requirements in NUREG-0737.

Also, in accordance with 10CFR50.54(m)(2), the shift complement during hot operations has been revised to require an additional operator with a Senior Reactor Operator's License.

Summary of Safety Analysis and Significant Hazards Determination

The change in STA staffing and shift manning makes the TS consistant with 10CFR50.54 and Sections I.A.1.1 and I.A.1.3 of NUREG-0737, therefore enhancing the technical ability of the shift complement.

This change constitutes additional restrictions on the shift complement not presently in Technical Specifications and it conforms to recent revisions in the regulations as stated. This change therefore does not involve a significant hazards consideration.

Based on the above, CP&L has determined that these changes do not constitute an unreviewed safety question, nor do they involve a significant increase in the probability or consequence 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 margin of safety.

Affected TS Page: 6.2-1

Revised Pages Attached: 6.2-1
6.2-1a

6.2 ORGANIZATION

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 CORE ALTERATIONS after the initial fuel 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.