3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

1. Cooldown and depressurize the RCS or

2. Heatup the RCS to above 350°F.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.4 Maximum Reactor Coolant Activity

The total specific activity in μCi/gram of the reactor coolant shall not exceed 1.0 μCi/gram dose equivalent I-131 and $100/\overline{E}$ μCi/gram under all modes of operation. ($\overline{E}$ is the average of beta and gamma energy (MEV) per disintegration of the specific activity.)

Whenever the reactor is critical or the average reactor coolant temperature is greater than 500°F, with the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity > 1.0 μCi/gram dose equivalent I-131 exceeding 500 hours in any consecutive six month period, prepare and submit a Special Report to the Commission within 30 days indicating the number of hours above this limit.

With the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.1.4-1, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

With the specific activity of the primary coolant > $100/\overline{E}$ μCi/gram, be in at least hot shutdown with T_{avg} < 500°F within 6 hours.

In any operating mode, with the specific activity of the primary coolant > 1.0 μCi/gram dose equivalent I-131 or > $100/\overline{E}$ μCi/gram, perform the sampling and analysis requirements of Item 1 of Table 4.1-2 until the specific activity of the primary coolant is restored to within its limits. A Reportable Event shall be reported pursuant to Specification 6.6. This report shall contain the results of the specific activity analyses together with the following information:

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal loops must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable.

a. If one residual heat removal loop becomes inoperable during cold shutdown operation, within 24 hours verify the existence of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method. Restore the inoperable RHR loop to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the loop to operable status.

b. If both residual heat removal loops becomes inoperable during cold shutdown operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, restore at least one residual

heat removal loop to operable status as soon as possible and follow the reporting requirements of Specifications 6.6.1 and 6.6.2.

3.3.2 Containment Cooling and Iodine Removal Systems

3.3.2.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 2505 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. Two containment spray pumps are operable.

c. Four fan cooler units are operable.

d. All essential features, including valves, controls, dampers, and piping associated with the above components are operable.

e. The system which automatically initiates the sodium hydroxide addition to the containment spray simultaneously to the actuation of the containment spray is operable.

3.4.2 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram dose equivalent I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is > 0.10 μCi/gram dose equivalent I-131, be in at least HOT SHUTDOWN within 6 hours and cold shutdown within the following 30 hours.

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

3.4.3 If, during power operations, any of the specifications in 3.4.1, with the exception of 3.4.1.b and 3.4.1.d as it applies to 3.4.1.b 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 an additional 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

3.4.4 With one auxiliary feedwater pump and/or essential features inoperable, restore that auxiliary feedwater pump and/or essential features to operable status within 72 hours, or;

a. Submit a Special Report to the Commission within 30 days outlining the cause of the inoperability and the action taken to return the pump and/or essential features to operable status, and;

b. Restore all three auxiliary feedwater pumps and their essential features to operable status within 7 days or be in at least hot shutdown within 6 hours.

3.4.5 With two auxiliary feedwater pumps inoperable, restore at least one inoperable auxiliary feedwater pump to operable status within 24 hours or be in at least hot shutdown within 6 hours.

TABLE 3.5-5
(THIS TABLE APPLIES WHEN THE RCS IS > 350°F)
INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

(HBR-06)

NO.	INSTRUMENT	1 MINIMUM CHANNELS OPERABLE	2 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 CANNOT BE MET
1	Pressurizer Level	2	See Item 9 Table 3.5-2
2	Auxiliary Feedwater Flow Indication (Primary Indication)		Note 1
	SD AFW Pump	1 per S/G	
	MD AFW Pump	1 per S/G	
3	Reactor Coolant System Subcooling Monitor	1	Note 2
4	PORV Position Indicator (Primary)	1	Note 3
5	PORV Blocking Valve Position Indicator (Primary)	1	Note 3
6	Safety Valve Position Indicator (Primary)	1	Note 3

Note 1: The three AFW lines from the MD AFW pumps and the three AFW lines from the SD AFW pump each contain one primary flow indicator (2 AFW flow paths per steam generator for a total of 6 AFW lines). These primary indicators are backed up by the narrow range steam generator level indications. If one or more of the direct AFW flow indicators becomes inoperable when the RCS is > 350°F, restore the indicator(s) to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause(s) of the inoperable indicator(s), the actions being taken to restore the indicator(s) to an operating status, the estimated date for completion of the repairs, and any compensatory action being taken while the indicator(s) is inoperable. The action required when any of the back up indications of AFW flow are inoperable, is described in Table 3.5-2.

Note 2: If both channels of the RCS subcooling monitor become inoperable when the RCS is >350°F, restore at least one channel to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause(s) of the inoperable channels, the actions being taken to restore at least one channel to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while both channels are inoperable.

Note 3: The Pzr PORVs and Pzr PORV blocking valves both incorporate limit switches for the direct (primary) means of position indication. The back up method of position indication consists of PRT pressure and a temperature element in a common line downstream of the valves. The Pzr safety relief valves incorporate a vibration monitoring system as the primary method of valve position indication. The back up method of position indication consists of a temperature element downstream of each valve and PRT pressure. If the primary method of position indication for either the Pzr PORVs, Pzr PORV blocking valves, or Pzr safety relief valves becomes inoperable when the RCS is > 350°F, restore the primary method to an operable status within 7 days, or prepare and submit a Special Report to the NRC within the following 14 days detailing the cause of the inoperable primary position indication method, the actions being taken to restore it to an operable status, the estimated date for completion of the repairs, and any compensatory action being taken while the primary position indication method is inoperable. If any of the back up methods of position indication for these valves becomes inoperable, it is to be repaired as soon as plant conditions permit.

e. Both batteries and the D.C. distribution systems are operable.

3.7.2 During power operation the following components may be inoperable:

a. Provided both diesel generators are operable, power operation may continue with the start-up transformer out of service for 24 hours without reporting to the NRC.

b. Power operation may continue with the start-up transformer out of service beyond 24 hours provided both diesel generators are operable and the reporting requirements of Specification 6.6.1 are followed.

c. Power operation may continue if the start-up transformer and one diesel generator is inoperable provided the reporting requirements of Specifications 6.6.1 and 6.6.2 are followed.

d. Power operation may continue for seven days if one diesel generator is inoperable provided the remaining diesel generator is tested daily to ensure operability and the engineered safety features associated with this diesel generator shall be operable.

3.14 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the operating status of the fire detection instrumentation, fire suppression systems, fire barriers, and to the administrative controls required for a comprehensive fire protection and prevention program. The requirements of these specifications shall apply to an area or areas when equipment in that area or areas is required to be operable as specified by other Limiting Conditions for Operation.

Objectives:

To assure the operability of Fire Protection Systems.

Specification:

3.14.1 Fire Detection and Actuation Instrumentation

3.14.1.1 As a minimum, the fire detection and actuation instrumentation for each fire detection zone shown in Table 3.14.1 shall be operable.

3.14.1.2 With the number of operable fire detection and actuation instruments less than required by Table 3.14.1:

a. For Fire Zones 24, 25A, 25B, 25C and 26 (inside Reactor Containment) initiate an inspection once per shift of the affected zone with particular emphasis on identifying any potential hazards for fire.

b. For all other fire zones, within one (1) hour increase the inspection frequency of the zone with the inoperable instrument(s) to at least once per hour.

c. Restore the inoperable instrument(s) to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrument(s) to operable status.

Basis

Operability of the fire detection and actuation instrumentation ensures that adequate warning capability is available for prompt detection of fires and provides for the actuation of automatic isolation and suppression systems which protect various safety related areas of the plant. The capabilities are required in order to detect, locate, isolate and extinguish fires in their early stages. Prompt detection of fires will reduce the potential for

damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection and actuation instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3.14.2 Fire Suppression Water System

3.14.2.1 The Fire Suppression Water System shall be operable with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the yard loop, and

b. An operable flow path capable of taking suction from the Unit 2 intake structure and transferring the water through distribution piping with operable sectionalizing, or isolation valves.

3.14.2.2 With less than the above required equipment operable:

Restore the inoperable equipment to operable status within seven days or prepare and submit a Special Report to the Commission within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3.14.2.3 With no Fire Suppression Water System operable:

a. Establish a backup fire suppression water system within 24 hours and follow the reporting requirements of Specification 6.6.1, or

b. Proceed to hot shutdown within twelve hours and be in cold shutdown within the next 24 hours.

3.14.3 Fire Water Pre-Action System

3.14.3.1 The Fire Water Pre-Action Systems in the first floor Auxiliary Building hallway above the instrument and service air compressor and the Containment Vessel Electrical Penetration Area shall be operable:

a. With no visible water leakage from the spray nozzles,
b. With the air supply to the system operable,
c. With automatic initiation logic operable, and
d. With the system aligned to deliver to the protection area.

3.14.3.2 With the Fire Water Pre-Action Systems in a condition of readiness less than required by the above:

a. For the Containment Vessel Electrical Penetration Area initiate an inspection once per shift with particular emphasis on identifying any potential hazards for fire.

b. For all other areas, within one (1) hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

c. Restore the system to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

3.14.4 <u>Fire Hose Stations</u>

3.14.4.1 Each fire hose station in Table 3.14.2 shall be operable.

3.14.4.2 With a hose station in Table 3.14.2 inoperable:

a. Route an additional equivalent capacity hose to the unprotected area from an operable hose station within one hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. Restore the hose station to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

3.14.5 <u>CO_2 Fire Protection System</u>

3.14.5.1 The CO_2 Fire Protection Systems for 1) the Diesel Generator Rooms and, 2) North and South Cable Vaults shall be <u>operable</u>, each:

a. With a complete bank (19 cylinders for the Diesel Generator Room and 18 cylinders in the North and South Cable Vaults) of fully charged CO_2 cylinders in service,

b. With the system aligned to deliver to the protected areas, and

c. With automatic initiation logic operable. For the Diesel Generators, this includes two dedicated heat detectors per room for CO_2 actuation.

d. A CO_2 cylinder shall be deemed fully charged if it contains not less than 90% of the full charge weight.

3.14.5.2 With any of the CO_2 Fire Protection Systems in a condition of readiness less than required by the above:

a. Within one (1) hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. Restore the affected system to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

c. If a diesel generator CO_2 Fire Protection System is inoperable and the affected diesel generator is running, immediately post a continuous fire watch. A continuous fire watch shall be maintained until the CO_2 fire protection system is restored to operability or until the diesel generator has been shut down.

3.14.6 Halon Fire Protection System

3.14.6.1 The Halon Fire Protection System for the Cable Spread Room Emergency Switchgear Room and the Safeguards Room shall be operable:

a. With a complete bank (10 cylinders, 5 instantaneous and 5 extended discharge) of fully charged Halon cylinders in service.

b. With the systems aligned to deliver to the protected areas.

c. With automatic initiation logic operable.

d. A Halon Cylinder shall be deemed to be fully charged if it contains not less than 90% of its full charge pressure and not less than 95% of its full charge weight.

3.14.6.2 With the Halon Fire Protection System in a condition of readiness less than required by the above:

a. Within one (1) hour establish a continuous fire watch with fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. Restore the system to operable status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

Basis:

The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where

safety related equipment is located. The fire suppression system consists of the water system, CO_2, Halon, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for immediate notification to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3.14.7 Fire Barrier Penetration Fire Seals

3.14.7.1 All penetration fire barriers protecting safety related areas shall be operable when equipment in those areas are required to be operable.

3.14.7.2 With the penetration fire barrier inoperable:

a. The operability of the fire detection systems providing coverage for the fire areas on either side of the penetration, as applicable, shall be verified within one hour.

b. If either of the detection systems are inoperable, a continuous fire watch shall be established on at least one side of the affected penetration within one hour.

c. Restore the inoperable fire barrier penetration(s) to operable status within 7 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperable penetration and plans and schedule for restoring the fire barrier penetration(s) to operable status.

Basis

The operability of the fire barrier penetration seals ensure that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetration seals are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not operable, verification of fire detection system operability is required to insure that prompt detection capability exists in the vicinity of the penetration barrier. Should an area detection system be inoperable, the fire watch will provide the required protection until the seal is restored to operable status.

3.15 CONTROL ROOM FILTER SYSTEM

Applicability

Applies to the Control Room filter system which is required for the safe operation of the plant. This system incorporates both HEPA filters and a charcoal adsorber bank.

Objective

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

Specification

3.15.1 During all modes of operation, except cold shutdown, the Control Room filter system shall be capable of performing its intended function in the required manner, except as described below:

a. If the system is determined to be inoperable, it shall be returned to operable status within seven days or prepare a Special Report which shall be submitted to the Commission within the next 14 days. This report shall outline the cause of the inoperability, the corrective actions taken, and the plans and schedule for restoring the system to an operable status.

3.15.2 If the system is determined to be inoperable while the reactor is in cold shutdown, the system shall be made operable prior to reactor startup.

Basis

Operability of the Control Room filter system ensures that the Control Room will remain habitable during an accidental atmospheric radiation release to

the extent that none of the occupants would receive a personnel radiation exposure in excess of 10 percent of the suggested limits in 10CFR100[1]. Because the system's protection is required only during low probability events, the system may be out of service for 7 days for repairs. Following this period, a Special Report detailing the status of the system will be submitted to the Commission. Since reactor startup should not commence without this system in service, the specification prohibits startup with the system inoperable.

(1) FSAR Section 6.4

(e) Unscheduled inspections shall be conducted in accordance with Specification 4.2.5.1.2 on any steam generator with primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) exceeding Specification 3.1.5.3.

All steam generators shall be inspected before returning to power in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineering safeguards, or a main steam line or feedwater line break.

4.2.5.1.5 Acceptance Limits

Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation equal to or greater than 20% of the nominal tube wall thickness.

Defect is an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which a degraded tube must be removed from service by plugging, because the tube may become defective prior to the next scheduled inspection of that tube. The plugging limit is 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the plugging limit for each 12 month period until the next inspection of the inspected steam generator.

4.2.5.2 Corrective Measures

All tubes that leak or are determined to have degradation exceeding the plugging limit shall be plugged prior to return to power.

4.2.5.3 Reports

1. After each inservice examination, the number of tubes plugged in each steam generator shall be reported to the Commission.

2. The complete results of the steam generator tube inservice inspection shall be included in the Operating Report for the period in which the inspection was completed.

Reports shall include:

(a) Number and extent of tubes inspected

(b) Location and percent of wall thickness penetration for each eddy current indication and any leaks

(c) Identification of tubes plugged.

3. All results in Category C-3 of Table 4.2.2 shall be reported to the Commission prior to resumption of plant operation. The written follow-up shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Basis:

The inspection program, where practical, is in compliance with Section XI of the ASME Code for In-service Inspection of Nuclear Reactor Coolant Systems dated January, 1970. Though examinations in certain areas are desirable, it should be recognized that equipment and techniques to perform the inspection are still in development. In all areas scheduled for volumetric examination, a detailed pre-service mapping will be conducted using techniques anticipated to be used for post-operation examinations. The areas indicated for inspection represent those of representative stress levels and therefore will serve to indicate potential problems before significant flaws develop there or at other areas. As more experience is gained in operation of pressurized water reactors, the time schedule and location of inspection may be altered or, should new techniques be developed, consideration may be given to incorporate these new techniques into this inspective program.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most primary loop components except the reactor vessel. The reactor vessel presents special problems because of the radiation levels and the requirement for remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps[1] have been incorporated into the design and manufacturing procedures in preparation for nondestructive test techniques which may be available in the future.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes for evidence of mechanical damage or progressive degradation. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects will be minimized with proper chemistry treatment of the secondary coolant. If defects or significant degradations should develop in service, this condition is expected to be detected during inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections by means of eddy current testing have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications.

References

(1) FSAR, Section 5.3.1

TABLE 4.2-2 (HBR-06)

STEAM GENERATOR TUBE INSPECTION
H. B. ROBINSON UNIT NO. 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (SG) S=3(N/n)% where: N is the number of steam generators in the plant = 3 n is the number of steam generators inspected during an examination	C-1	Acceptable for Continued Service	N/A	N/A	N/A	N/A
	C-2	Plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
			C-2	Plug tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
					C-2	Plug tubes exc. plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect all tubes in this SG, plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in each other steam generator not included in the inservice inspection program. Report results to NRC.	All other SGs are C-1	Acceptable for Continued Service	N/A	N/A
			Some SGs are C-2 but no additional SGs are C-3	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			Additional SG is C-3	Inspect all tubes in SG and plug tubes exceeding the plugging limit. Report results to NRC.	N/A	N/A

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a Special Report to the Commission within 30 days.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should

6.5.1.2.4 Modifications that are determined to either constitute an unreviewed safety question, as defined in 10CFR50.59(a)(2), or a change to the Technical Specifications, shall be reviewed by the Plant Nuclear Safety Committee and submitted to the NRC for approval prior to implementation. All such modifications shall be approved by the Corporate Nuclear Safety Section prior to implementation.

6.5.1.2.5 Modifications which constitute changes to the facility as described in the FSAR shall also be reviewed by the Corporate Nuclear Safety Section. This review may be conducted after plant management approval, and implementation may proceed prior to completion of review.

6.5.1.3 <u>Technical Specifications and License Changes</u>

6.5.1.3.1 Each proposed Technical Specification or Operating License change shall be reviewed by the Plant Nuclear Safety Committee and submitted to the NRC for approval.

6.5.1.4 <u>Review of Technical Specification Violations</u>

6.5.1.4.1 All violations of Technical Specifications shall be investigated and a report prepared that evaluates the event and that provides recommendations to prevent recurrence. Such reports shall be reviewed by the PNSC and approved by the Plant General Manager or has designee and submitted to the Vice President - Nuclear Operations and to the Manager - Corporate Nuclear Safety.

6.5.1.5 <u>Nuclear Safety Review Qualification</u>

6.5.1.5.1 Individuals shall be designated by the Plant General Manager for the safety reviews of Specifications 6.5.1.1.2, 6.5.1.1.3, 6.5.1.2.1, and 6.5.1.2.2. These reviewers shall have a Bachelor of

6.5.1.6.5 A quorum of the PNSC shall consist of the Chairman, and three members, of which two may be alternates.

6.5.1.6.6 The PNSC activities shall include the following:

a. Perform an overview of Specifications 6.5.1.1 and 6.5.1.2 to assure that processes are effectively maintained.

b. Performance of special reviews, investigations, and reports thereon requested by the Manager - Corporate Nuclear Safety.

c. Annual review of the Security Plan and Emergency Plan.

d. Perform reviews of Specifications 6.5.1.1.6, 6.5.1.2.4, 6.5.1.3.1, and 6.5.1.4.1.

e. Perform review of all reportable events.

f. Review of facility operations to detect potential nuclear safety hazards.

6.5.1.6.7 In the event of disagreement between the recommendations of the Plant Nuclear Safety Committee and the actions contemplated by the General Manager, the course determined by the General Manager to be more conservative will be followed. The Vice President - Nuclear Operations and the Manager - Corporate Nuclear Safety will be notified within 24 hours of the disagreement and subsequent actions.

6.5.1.6.8 The PNSC shall maintain written minutes of each meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications; and copies shall be provided to the Vice President - Nuclear Operations, and to the Manager - Corporate Nuclear Safety.

6.5.2 Corporate Nuclear Safety Section - Independent Review

The Corporate Nuclear Safety Section of the Corporate Nuclear Safety & Research Department shall provide independent review of significant plant changes, tests, and procedures; verify that reportable events are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer. Specific review subjects are defined in Specification 6.5.2.1.d.

6.5.2.1 The Manager - Corporate Nuclear Safety, under the Vice President - Corporate Nuclear Safety & Research, is charged with the overall responsibility for administering the independent review function as follows:

a. Approves selection of the individuals to conduct safety reviews under Specification 6.5.2.

b. Has access to plant records and operating personnel in performing independent reviews.

c. Prepares and retains written records of reviews.

d. Assures independent reviews are conducted on the following subjects:

(1) Written safety evaluations of changes in the facility as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report, and tests or experiments not described in the Safety Analysis Report that are completed without prior NRC approval under the provisions of 10CFR50.59(a)(1). This review is to verify that such changes, tests, or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as

defined in 10CFR50.59(a)(2). These reviews may be conducted after appropriate management approval, and implementation may proceed prior to completion of the review.

(2) Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an unreviewed safety question pursuant to 10CFR50.59(c). Matters of this kind shall be referred to the Corporate Nuclear Safety Section by the Plant General Manager or by other functional organizational units within Carolina Power & Light Company prior to implementation.

(3) Proposed changes to the Technical Specifications or this operating license, prior to implementation.

(4) All reportable events.

(5) Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section, deems appropriate for consideration of which is referred to the Manager - Corporate Nuclear Safety Section, by the on-site operating organization or by other functional organizational units within Carolina Power & Light Company.

(6) Reports and minutes of the PNSC.

6.5.2.2 Results of Corporate Nuclear Safety reviews, including recommendations and concerns, shall be documented.

a. Copies of documented reviews shall be retained in the CNS files.

b. Recommendations and concerns shall be submitted to the plant General Manager and Vice President - Nuclear Operations, within 14 days of determination.

c. A summation of Corporate Nuclear Safety recommendations and concerns shall be submitted to the Chairman/President; Vice Chairman; Executive Vice President - Power Supply and Engineering & Construction; Senior Vice President - Power Supply; Vice President - Nuclear Operations; Vice President - Nuclear Safety & Research; plant General Manager; and others, as appropriate on at least a bimonthly frequency.

d. The corporate Nuclear Safety Review program shall be conducted in accordance with written, approved procedures.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for events requiring immediate notification:

a. The NRC shall be notified pursuant to the requirements of 10CFR50.72.

b. Each reportable event shall be reviewed in accordance with Specification 6.5.1.6.6 and submitted to the Manager - Corporate Nuclear Safety Section, and the Vice President - Nuclear Operations.

6.6.2 The following actions shall be taken for reportable events requiring a Licensee Event Report:

a. A report shall be submitted to the NRC pursuant to the requirements if 10CFR50.73.

b. Each reportable event shall be reviewed in accordance with Specification 6.5.1.6.6 and submitted to the Manager - Corporate Nuclear Safety Section, and the Vice President - Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10CFR50.72 shall be complied with.

b. The provisions of 10CFR50.36(c)(1)(i) shall be complied with.

c. The Safety Limit violation shall be reported to the NRC Region II within one hour and to the Vice President - Nuclear Operations and the Manager - Corporate Nuclear Safety Section within 24 hours.

d. A Safety Limit Report shall be prepared. The report shall be reviewed in accordance with Specification 6.5.1.6.6. This report shall describe (1) applicable circumstances preceding the violation; (2) effects of the violation upon facility components, systems, or structures; and (3) corrective action taken to prevent recurrence.

e. The Safety Limit Violation Report shall be submitted to the NRC Region II, Vice President - Nuclear Operations, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

c. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis. The report formats set forth in Appendices B, C, and D to Regulatory Guide 1.16 shall be completed in accordance with the instructions provided. The completed forms should be submitted by the tenth of the month following the calendar month covered by the report to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy of the appropriate NRC Regional Office.

6.9.2 Deleted

6.9.3 Special Reports

Special reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

	Area	Reference	Submittal Date
a.	Containment Leak Rate Testing	4.4	Upon completion of each test
b.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 25 years of operation
c.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 20 years of operation

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of facility operation covering time interval at each power level.

b. Records of principal maintenance activities, inspections, repair and replacement of principal items of equipment, related to nuclear safety.

c. Reportable Event Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak test and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.