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SCHWENCER,A. OPERATING REACTORS BRANCH 1

SUBJECT: REQUESTS REVISIONS OF TECH SPECS TO FACILITATE WORK PLANNED FOR OUTAGE TO CLARIFY OR CHANGE TECH SPECS TO BE IN ACCORDANCE W/NRC GUIDELINES.PROPOSED CHANGES & SAFETY EVALUATION ENCL.REQUESTS ISSUANCE BY 790412.

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Carolina Power & Light Company

March 6, 1979

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Office of Nuclear Reactor Regulation
ATTENTION: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
United States Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

Dear Mr. Schwencer: REGULATORY DOCKET FILE COPY

In accordance with the Code of Federal Regulations, Title 10, Part 50.90 and Part 2.101, Carolina Power & Light Company hereby requests revisions to the Technical Specifications for its H. B. Robinson Steam Electric Plant, Unit No. 2. The changes discussed below concerning the removal of part length control rods and the blocking of the high differential pressure between any steamline and the steamline header signal during cooldown are required to facilitate work being planned for the upcoming refueling outage. The other changes clarify existing Technical Specifications or change them to conform with the most recent NRC guidance and references. Changes to the Technical Specifications are indicated by a vertical line in the right-hand margins of the affected pages, which are enclosed.

The enclosed changes to Section 3.10 and Section 5.3 reflect the Company's intention to remove the part length control rods, install thimble plugs, and place antirotation devices on the affected control rod drive mechanisms. The original intent of the part length rods was to provide the capability to position extra poison selectively in the core at areas of high neutron flux and thus provide increased operational margins and flexibility. Use of the part length rods, however, has been prohibited by the NRC, and all part length rods are required to be fully withdrawn from the core during reactor operation. During refueling, the part length rods are fully inserted. Their presence during refueling operations requires unnecessary operational and maintenance costs, extends the time requirements of the outage, and causes unnecessary occupational exposure with no beneficial results. Westinghouse Corporation, the NSSS vendor, has analyzed the modification and concluded that it provides no problems involving either safety analyses or core physics calculations. This modification has been made on several Westinghouse pressurized water reactors. A copy of Westinghouse's Safety Evaluation is attached.

The change to Table 3.5.3 is to clarify that the note allowing blockage of channels when the Reactor Coolant System (RCS) pressure is less than 2000 psig is applicable to Item 1c, "High Differential Pressure Between any Steamline and the Steamline Header." This note (identified by *** in the table) is applicable for both Item 1c and 1d, "Pressurizer Low Pressure and Low Level." Blockage of these channels is necessary during normal plant cooldown to prevent inadvertant activation of safety injection. Inadvertant activation of safety injection is highly undesirable because it interrupts the normal cooldown and has the potential

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for causing too rapid a cooldown and possible overpressurization. In actuality, the H. B. Robinson design is such that the high differential pressure between any steamline and the steamline header channel is blocked when the pressurizer low pressure and low level channel is blocked. These blocks are controlled by the same block switch and the latter is allowed when RCS pressure is less than 2000 psig. This change clarifies the table and makes the table agree with the installed design. A NRC regional inspector has requested that this change be made.

The changes to Sections 3.8.2b, 4.12, and Table 4.1-3 are to clarify the specifications. All technical changes conform with Regulatory Guide 1.52, Revision 2.

In accordance with 10CFR170.12(c), we have determined that this request constitutes two Class II amendments and one Class III amendment. Our check for $6,400 is enclosed as payment for this amendment fee.

It is requested that these changes be issued by April 12, 1979. The timely issuance of these amendments will facilitate the work planned for the refueling outage.

If your staff has any questions concerning the attached information, we will be glad to discuss them either by telephone or at a meeting with representatives of your staff.

Yours very truly,

E. E. Utley

E. E. Utley
Senior Vice President
Power Supply

JJS/mf
Enclosures

Sworn to and subscribed before me this 6th day of March, 1979

Margaret M. Cox
Notary Public

My Commission Expires: July 4, 1980

SAFETY EVALUATION

REMOVAL OF PART LENGTH CONTROL ELEMENT ASSEMBLIES

H. B. ROBINSON UNIT 2

I. INTRODUCTION

This report provides information to justify plant operation following the removal of the part-length rod cluster control assemblies (PLRCCA's). Plant operation at power is currently not allowed with PLRCCA's in the core.

Thimble plug assemblies will be installed into the locations previously occupied by the PLRCCA's. These plugs are being installed to preserve the current dynamic operating characteristics of the reactor, i.e., pressure drops, coolant flow rates, etc., which could be affected if just removal of the PLRCCA's was performed.

II. THIMBLE PLUG ASSEMBLY MECHANICAL DESIGN

The Thimble Plug Assembly, which will be inserted into locations previously occupied by PLRCCA's, consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the springs, are constructed from type 304 stainless steel. The springs are wound from Inconel X-750 for corrosion resistance and high strength.

These thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCA's have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, sources rods, or burnable poison rods.

III. THERMAL HYDRAULIC EFFECTS

A. Thermal Effects

Physics analysis, as well as incore monitoring, indicates that there will be no adverse effect of the plug assemblies on the core power distribution. Since the plugged fuel assemblies have no adverse effect on the design core flow distribution, calculated core thermal margin will be unaffected.

B. Hydraulic Effects

Hydraulic aspects were considered with respect to the installation of the thimble plug assemblies. Since the plug assemblies are already extensively used in existing fuel assemblies with no adverse effects, it can be concluded that there will be no adverse effects from the installation of these additional thimble plugs.

V. NEUTRONICS EFFECTS

The removal of part length rods has no impact on any physics information generated in the past for H. B. Robinson 2 Nuclear Plant. The use of part length RCCA's has been prohibited by Technical Specifications and they have been locked in the full out position during operation. The installation of thimble plug assemblies as described in Sections II and III will have no influence on the physics characteristics of the reactor. The lowest position of the plug assemblies will not be within several inches of the top of the fuel. Therefore, operation with installed plugs will not invalidate any of the physics parameters.

V. ACCIDENT AND TRANSIENT ANALYSES

Based on foregoing discussion, the following conclusions relating to accident and transient analyses can be reached.

A. Impact on Probability of Occurrence

A potential safety concern is that the probability of some event previously analyzed can be increased due to the replacement of PLRCCA's with thimble plug assemblies. No information exists which suggests that the replacement of PLRCCA's with thimble plug assemblies increases the probability of any event previously analyzed.

B. Other Malfunctions Not Previously Analyzed

No Information exists which suggests that the replacement of PLRCCA's with the thimble plug assemblies introduces a possibility for an accident or any malfunction of a different type than those previously analyzed. Hence, it is concluded that the replacement of PLRCCA's with plugs does not introduce the possibility of events not previously analyzed.

C. Margin of Safety

It is evaluated that the consequences of replacing the PLRCCA's with thimble plug assemblies does not reduce the margin of safety, as defined in the bases for applicable technical specifications.

D. Summary

The probability of occurrence of events has not increased and the consequences of these events remain within those reported in previous analyses. The possibility of other types of accidents or malfunctions has not increased. Hence, the information presented in this report leads to the conclusion that operation of H. B. Robinson 2 Nuclear Plant with the thimble plug assemblies instead of PLRCCA's does not present any danger to the health and safety of the public.

TABLE 3.5-3

INSTRUMENTATION OPERATING CONDITIONS FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1.	SAFETY INJECTION			
	a. Manual	1	0	Cold shutdown
	b. High Containment Pressure (Hi Level)	2	1	Cold shutdown
	c. High Differential Pressure between any Steam Line and the Steam Line Header	2	1	Cold shutdown***
	d. Pressurizer Low Pressure and Low Level	2*	1	Cold shutdown***
	e. High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} or Low Steam Pressure	1/Steam line	*****	Cold shutdown****
		2 T_{avg} Signals	1	
		2 Pressure Signals	1	
2.	CONTAINMENT SPRAY			
	a. Manual	2	0**	Cold shutdown
	b. High Containment Pressure (Hi-Hi Level)	2/set	1/set	Cold shutdown

*Each channel has two separate signals (level and pressure).
**Must actuate two switches simultaneously.
***When primary pressure is less than 2000 psig, channels may be blocked.
****When primary temperature is less than 547°F, channels may be blocked.
*****In this case the 2/3 high steam flow is already in the trip mode.

3.10.1.5 Except for physics tests, if a full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power for three-loop operation.

3.10.1.6 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted.

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.20/P) \times K(Z)$ for $P > .5$

$F_Q(Z) < (4.40) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} < 1.55\ (1 + 0.2(1-P))$

where P is the fraction of licensed power at which the core is operating, K(Z) is the function given in Figure 3.10-3, and Z is the core height location of F_Q.

3.10.2.1.2 The predetermined power level at which APDMS initiation is required is given by the relation

$$P_{APDMS} \leq \frac{1.435}{Fxy} \times 0.94$$

3.10.2.1.3 Fxy shall be determined for the unrodded core plane regions away from fuel support grids, located between a core plane elevation 2.0 feet from the top of the core and a core plane elevation 2.0 feet from the bottom of the core with no control rod inserted more than 2.0 feet into the core. This determination shall be made from the movable incore detector maps specified in 3.10.2.3.

3.10.2.2 If either measured hot channel factor exceeds these values the reactor power shall be reduced so as not to exceed a fraction of the design value equal to the ratio of the F_Q^N or $F_{\Delta H}^N$ limit to measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio. If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

3.10.2.3 Following initial loading and at regular monthly intervals thereafter, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied. For the purpose of this confirmation:

a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.4 Rod Drop Time

3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

3.10.5 Deleted

3.10.6 Inoperable Control Rods

3.10.6.1 A control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met.

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

3.10.6.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that shutdown margin equal to or greater than shown on Figure 3.10-2 results.

3.10.7 Power Ramp Rate Limits

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g., refueling, inspection), the rate of reactor power increase shall be limited to 3 percent of full power in an hour between 20 percent and 100 percent of full power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be removed for reactor power levels below a power level P (20 percent $<P \leq 100$ percent) provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-day operating period following the shutdown.

3.10.7.2 The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3 percent of full power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of full power followed by a maximum ramp rate of 3 percent of full power in an hour beginning three hours after the step increase.

3.10.8 Required Shutdown Margins

3.10.8.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.

shutdown margin. The specified control rod insertion limits meet the design basis criteria on (1) potential ejected control rod worth and peaking factor,[4] (2) radial power peaking factors, $F_{\Delta H}$, and (3) required margin shutdown.

The various control rod banks (shutdown banks, control banks) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks, and a linear position indicator (LVDT) which indicates the actual rod position.[2] The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because complete rod misalignment (control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.1.6) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal

area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

d. $F_{\Delta H}^{N}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^{N}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes through the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^{N}$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided the values of F_q and $F_{\Delta H}$ in Specification 3.10.2.1 are not exceeded.

For normal operation, it is not necessary to measure these quantities. Instead, it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows:

a. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

b. Control rod banks are sequenced with overlapping banks as shown in Figure 3.10-1.

c. The control bank insertion limits are not violated.

d. Deleted

e. Axial power distribution control procedures, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined on the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power, the design limits are met, provided the limits of Specification 3.10.2.1 are not exceeded.

The permitted relaxation in $F_{\Delta H}^{N}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset, it varies primarily with burnup.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with control Bank D more than 190 steps withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the specified diviation of ΔI is

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

Normal reactor operation causes significant pellet cracking and fragmentation. Consequently, handling of irradiated fuel assemblies can result in relocation of these fragments against the cladding. Calculations show that high cladding stresses can occur if the reactor power increase is rapid during the subsequent startup.

The 72-hour period allows for stress relaxation of the clad before the ramp rate requirement is removed, thereby reducing the potential harmful effects of possible pellet or fragment relocation.

The 3 percent limit is imposed to minimize the effects of adverse cladding stresses resulting from part power operation for extended periods of time. The time period of 30 days is based upon the successful power ramp demonstrations performed on Zircaloy clad fuel in operating reactors, resulting in no cladding failures.

References

(1) FSAR Section 14 and WCAP-8243
(2) FSAR Section 7.3
(3) WCAP-8243, Section 4.4.2
(4) WCAP-8243, Section 4.4.3

5.3.1.5 There are 45 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain 144 inch length of silver-indium-cadmium alloy clad with the stainless steel.[5]

5.3.1.6 Up to 10 grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the Reactor Coolant System complies with the Code requirements.[6]

5.3.2.2 All piping, components and supporting structures of the Reactor Coolant System are designed to Class I requirements.

5.3.2.3 The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 9343 cubic feet.[7]

References

(1) FSAR Section 3.2.3
(2) FSAR Section 3.2.1
(3) FSAR Section 3.2.1
(4) FSAR Section 3.2.3
(5) FSAR Sections 3.2.1 and 3.2.3
(6) FSAR Table 4.1-9
(7) FSAR Table 4.1-1
(8) "Description and Evaluation of Test Assemblies Containing Gadolinia Bearing Fuel Rods" submitted with letter dated January 5, 1973, from CP&L to the Director of Licensing.
(9) "Description and Evaluation of Test Assemblies Containing Gadolinia Bearing Fuel Rods - H. B. Robinson Unit No. 2 Cycle 3" submitted with letter dated March 12, 1974, from CP&L to the Director of Licensing.

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 adsorber banks shall show $\geq$99 percent DOP removal and $\geq$99 percent halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon shall show $\geq$90 percent radioactive methyl iodide removal in accordance with test 5.b of Table 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 $\pm$10% of 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.

TABLE 4.1-3 (Continued)

		Check	Frequency	Maximum Time Between Tests
13.	Turbine Inspection	Visual, Magnaflux and Die Penetrant	Every five years	6 years
14.	Fans and Associated Charcoal and Absolute Filters for Control Room and Residual Heat Removal Compartments (HVE-19, HVE-5a and 5b respectively)	Fans functioning. Laboratory tests on charcoal must show $\geq$ 99% iodine removal. In-place test must show $\geq$ 99% removal of polydispersed DOP particles by the HEPA filters and Freon by the charcoal filters.	Once per operating cycle.	NA
15.	Isolation Seal Water System	Functioning	Each refueling shutdown	NA

*NA - Not applicable

4.12 REFUELING FILTER SYSTEMS

Applicability

Applies to fans and associated charcoal adsorber banks and HEPA filters for Spent Fuel Building filter system and Containment Purge filter system.

Objective

To verify that the refueling filter systems will adequately remove radioactivity that may be released accidentally into the Spent Fuel Building and Containment Building.

Specification

4.12.1 At least once per operating cycle, the following conditions shall be demonstrated:

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate.

b. Fan capacity shall be within $\pm$ 10% of the design flow.

4.12.2 a. The tests of Specification 3.8.2.a for the refueling filter systems shall be performed initially, and at least once per operating cycle prior to each refueling outage operation or after every 720 hours of system operation whichever occurs first.

b. The tests and sample analysis of Specification 3.8.2.b for the refueling filter systems shall be performed initially, at least once per operating cycle prior to

each refueling outage operation or after every 720 hours of system operation, whichever occurs first, and following significant painting, fire, or chemical release in any ventilation zone communicating with the filter system.

c. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance of the filter system housing.

d. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the filter system housing.

e. A uniform air distribution within ± 20% across HEPA filters and charcoal adsorbers must be demonstrated initially and after each major repair or modification to the systems which would affect the air distribution.

4.12.3 The relative humidity of the air processed by the refueling filter system shall be monitored hourly during fuel handling operations.

Basis

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop and fan capacity should be determined a least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated under postulated accident conditions. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

with an absorbent qualified according to Table 5.1 of ANSI/ASME N509-1976. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

The Containment Purge filter system is normally run continuously during the entire refueling outage to provide cooling and ventilation and periodically during plant operation to reduce airborne radioactivity leaks inside the containment. Operation time of the Containment Purge filter system after the fuel handling operation is completed should not be added to the operation time during fuel handling operations for determination of testing and surveillance requirements given in these specifications.

If significant painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign material, the same laboratory tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

The relative humidity of the Containment atmosphere and air downstream of the heaters in the Spent Fuel Building filter system shall be monitored at least hourly to assure that the R.H. is less than 70 percent during fuel handling and Containment Purge filter system operation.