July 20, 1999

Docket File

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SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - TECHNICAL SPECIFICATION BASES CHANGE (TAC NO. MA5510)

Dear Mr. Gambhir:

On July 15, 1999, the NRC staff issued revised technical specification Bases that were provided by your letter of September 4, 1998, for the Fort Calhoun Station Technical Specifications. Due to an administrative oversight these pages were not dated. Enclosed are dated pages. We apologize for any inconvenience this may have caused.

Sincerely,

/s/

L. Raynard Wharton, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure: Bases Pages

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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1.3 Limiting Safety System Settings, Reactor Protective System (continued)

(7) Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment high pressure signal which indicates safety injection system operation.

(8) Axial Power Distribution - The axial power trip is provided to ensure that excessive axial peaking will not cause fuel damage. The Axial Shape Index is determined from the axially split excore detectors. The set point functions, shown in the COLR ensure that neither a DNBR of less than 1.18 nor a maximum linear heat rate of more than 22 kW/ft (deposited in the fuel) will exist as a consequence of axial power maldistributions. Allowances have been made for instrumentation inaccuracies and uncertainties associated with the excore symmetric offset - incore axial peaking relationship. A variance of 5% between ΔT-Power and NI-Power is permitted due to the significant margins to local power density limits before calibration of NI-Power is performed at 30% power.

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than 1.18 nor a peak linear heat rate of more than 22 kW/ft occurs as a result of the loss of load to one steam generator.

(10) Physics Testing at Low Power - During physics testing at power levels less than 10^{-1}% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below 10^{-1}% of rated power. Written test procedures which are approved by the Plant Review Committee will be in effect during these tests. At reactor power levels less than 10^{-1}% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6.2.2, 14.6.4
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6
(9) USAR, Section 14.10

2.0.1 General Requirements (Continued)

function and have at least one normal and one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 2.7(1)b requires that both House Service Transformers T1A-3 and T1A-4 be OPERABLE. Specification 2.7(2)c provides a 72-hour out-of-service time when both required House Service Transformers T1A-3 and T1A-4 are not OPERABLE provided the operability of both Diesel Generators is immediately verified. If the definition of OPERABLE were applied without consideration of Specification 2.0.1(2), all systems, subsystems, trains, components, and devices supplied by the inoperable House Service Transformers T1A-3 and T1A-4 would also be inoperable. This would dictate invoking the applicable measures for each of the applicable LCO's. However, the provisions of Specification 2.0.1(2) permit the time limits for continued operation to be consistent with the corrective measures for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components, and devices in both divisions must be also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

In Operating Modes 4 or 5, Specification 2.0.1(2) is not applicable, and thus the individual requirements for each applicable Limiting Condition for Operation in these modes must be adhered to.

Action statements (5)b. and c. include the removal of power from a closed block valve to preclude any inadvertent opening of the block valve at a time the PORV may not be closed due to maintenance. However, the applicability requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling shutdown (Mode 5), so that maintenance can be performed on the PORV(s) to eliminate the seat leakage condition.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the main steam safety valves. Conservative values for all systems parameters, delay times and core moderator coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half of the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten main steam safety valves is 6.606×10^6 lb/hr. If, following testing, the as found setpoints are outside +/-1% of nominal nameplate values, the valves are set to within the +/-1% tolerance. The main steam safety valves were analyzed for a total loss of main feedwater flow while operating at 1500 MWt[3] to ensure that the peak secondary pressure was less than 1100 psia, the ASME Section III upset pressure limit of 10% greater than the design pressure. At the power of 1500 MWt, sufficient relief valve capacity is available to prevent overpressurization of the steam system on loss-of-load conditions.[4] These analyses are based on a minimum of four-of-five operable main steam safety valves on each main steam header.

The power-operated relief valve low setpoint will be adjusted to provide sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3, to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor. The effective full flow area of an open PORV is 0.94 in^2.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III
(2) USAR, Section 14.9
(3) USAR, Section 14.10
(4) USAR, Sections 4.3.4, 4.3.9.5

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 385°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

(4) Trisodium Phosphate (TSP) Dodecahydrate

During operating Modes 1 and 2, the TSP baskets shall contain ≥ 110 ft^3 of active TSP.

a. With the above TSP requirements not within limits, the TSP shall be restored within 72 hours.

b. With Specification 2.3(4)a required action and completion time not met, the plant shall be in hot shutdown within the next 6 hours and cold shutdown within the following 36 hours.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical. The energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. |

Basis (Continued)

The SIRW tank contains a minimum of 283,000 gallons of usable water containing a boron concentration of at least the refueling boron concentration. This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 68°F.[(2)]

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3. Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and therefore in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to assure that operability of the component will be restored promptly and yet allow sufficient time to effect repairs using safe and proper procedures.

The time allowed to repair a safety injection tank is based on the deterministic and probabilistic analyses of CE NPSD-994, "CEOG Joint Applications Report for Safety Injection Tank AOT/SIT Extension," May 1995. These analyses concluded that the overall risk impact of the completion times are either risk-beneficial or risk neutral.

The requirement for core cooling in case of postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition reduces the consequences of a loss-of-coolant accident and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition.

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.(3)(4)

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for both large break analysis and small break analysis it is assumed that one high pressure pump and one low pressure pump operate(5); and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown in USAR Section 14.

The restriction on HPSI pump operability at low temperatures, in combination with the PORV setpoints ensure that the reactor vessel pressure-temperature limits would not be exceeded in the case of an inadvertent actuation of the operable HPSI and charging pumps.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path. The HPSI pump flow rate must be restricted to that of three charging pumps in order to minimize the consequences of a mass addition transient while at low temperatures.

Trisodium Phosphate (TSP) dodecahydrate is required to adjust the pH of the recirculation water to ≥ 7.0 after a loss of coolant accident (LOCA). This pH value is necessary to prevent significant amounts of iodine, released from fuel failures and dissolved in the recirculation water, from converting to a volatile form and evolving into the containment atmosphere. Higher levels of airborne iodine in containment may increase the releases of radionuclides and the consequences of the accident. A pH of ≥ 7.0 is also necessary to prevent stress corrosion cracking (SCC) of austenitic stainless steel components in containment. SCC increases the probability of failure of components.

The required amount of TSP is represented in a volume quantity converted from the Reference 7 mass quantity using the manufactured density. Verification of this amount during surveillance testing utilizes the measured volume.

to function during accident conditions may be inoperable for a period of no more than 24 hours. If operability is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours.

Basis

A full capacity diesel-generator is connected to each of the two engineered safeguards 4.16-kV buses. Three engineered safeguards 480-Volt double-ended load centers are provided; of the six transformers, three are connected to each of the two 4.16-kV buses. Two load centers are operated as two-bus-section units; the third is provided with a center bus manually transferable to either associated end section. The center bus section supplies HPSI Pump SI-2C, CS Pump SI-3C and Charging Pump CH-1C any of which can thus be supplied from either 4.16-kV bus if required. The containment sprays initially take coolant from the safety injection and refueling water (SIRW) tank. Before this supply of water is exhausted (at least 24 minutes)(2) the spray system is transferred to the recirculation mode and the pumps take suction from the containment sump. One shutdown cooling heat exchanger is sufficient to satisfy the spray system requirements during the long-term containment cooling period.(3) In addition, in the unlikely event of the component cooling water supply being lost, raw water can be utilized for direct cooling of certain engineered safeguard components.(4)

The containment spray system is independent from the containment air recirculation, cooling and iodine removal system for the containment cooling function.(5) For the limiting Loss of Coolant Accident (LOCA) scenario, one of the three spray pumps would limit the containment pressure to below the design value without taking credit for the air coolers or the cooling capacity of the safety injection system.(6)(7) For the limiting Main Steam Line Break (MSLB) scenario, a heat removal contribution is credited from the air coolers in the mitigation of containment peak pressure.(7)

The cooling equipment provided to limit the containment pressure following a DBA is divided between the independent power supply systems. The raw water and component cooling water pumps are similarly distributed on the 4.16-kV and 480 Volt buses. In the event of a DBA, loss of normal power sources and failure of one diesel-generator to operate, a minimum of at least one spray pump, and two air coolers would be connected to the available diesel-generator. This would provide adequate containment cooling equipment to limit the containment pressure below the design value for the limiting one pump, one spray header LOCA event. The limiting MSLB event in which off site power is available, is not affected by the loss of one diesel generator.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

The component cooling system pumps and heat exchanger, the spray pumps and the shutdown heat exchangers are located in the auxiliary building.[9][10] The raw water pumps are located in the intake structure.[11]

Analyses show that after a high heat load accident such as a large break LOCA or Main Steam Line Break inside containment, three in service component cooling heat exchangers will maintain CCW return temperature in an analyzed range. This assumes all of the containment air cooling units are operating which would create the maximum heat load on the CCW system. In order to ensure that three heat exchangers would be in service after a DBA in conjunction with an assumed single failure, four are required to be operable.

If the river temperature is below 70°F, the single failure of a component cooling heat exchanger's RW valve to open would not raise CCW return temperature to an unanalyzed level. A single failure of a RW valve to auto-open will raise CCW return temperature due to CCW passing through a heat exchanger which has no flow. If the river temperature is greater than or equal to 70°F, a single failure of a heat exchanger's RW valve to open could raise CCW return temperature to an unanalyzed level. Therefore, when the river temperature is greater than or equal to 70°F and RW is not in service to a heat exchanger, the heat exchanger is considered inoperable. Having RW in service eliminates the potential failure of a RW valve to open as a credible single active failure.

When a component cooling heat exchanger is inoperable for maintenance and the river temperature is greater than or equal to 70°F, at least one RW and CCW isolation valve is secured in the closed position. This prevents the valve from opening upon receipt of a Safety Injection Actuation Signal, and therefore prevents CCW passing through a heat exchanger that has no RW flow. If the opening stroke time of a RW or CCW isolation valve on a heat exchanger is such that the valve is considered inoperable, the heat exchanger associated with the inoperable valve is still operable as long as the affected RW or CCW valve is maintained in the open position. The RW and CCW isolation valves may be opened or closed intermittently under administrative control without entering the LCO action statement in order to conduct required testing of components.

References

(1) Deleted

(2) USAR, Section 6.2.3.1

(3) USAR, Section 6.2.3.4

(4) USAR, Section 9.8.2

(5) USAR, Section 6.4.5

(6) USAR, Section 6.3.5

(7) USAR, Section 14.16.5

(8) Deleted

(9) USAR, Section 9.7

(10) USAR, Section 6.3

(11) USAR, Section 9.8

2.0 LIMITING CONDITIONS FOR OPERATION

2.13 Nuclear Detector Cooling System

Applies to the operational status of the Nuclear Detector Cooling System.

Objective

To assure that the concrete in the biological shield surrounding the reactor vessel does not overheat.

Specifications

(1) The annulus exit temperature from the nuclear detector cooling system shall not exceed a temperature found to correlate to 150°F concrete temperature.

(2) There will be at least 2 temperature detectors in service to measure annulus exit air temperatures whenever the reactor is in service.

Basis

The Nuclear Detector Cooling System is used to cool the air in the annulus between the reactor vessel and the biological shield. While the nuclear detectors can withstand temperatures considerably higher than 150°F the elevated temperature could result in reduction in concrete strength through loss of moisture. Each nuclear detector well cooling unit is rated at 100% of the system design capability of 173,00 Btu/hr.[1]

A test was performed during Hot Functionals and/or Low Power Tests to determine (1) the correlation between annulus air temperature and concrete temperature; (2) rate at which the concrete will heat up if no cooling is available.

The results of these tests were used to provide control room indication of concrete temperatures (that is annulus air temperature) and allowable reactor operation time in the event both nuclear detector well cooling units were inoperable.

The objective for this specification is to hold the concrete bulk temperature to no greater than 150°F. The annulus exit temperature which correlated to 150°F (concrete temperature) was determined by testing. Temperature sensors are installed in the concrete and in the annulus air discharge. The sensors in the concrete are subjected to neutron flux during operation and are no longer functional. The value determined for annulus exit temperatures which correlate with concrete temperatures were determined and a maximum value used to comply with the Tech Spec limit.

References

(1) FSAR, Section 9.10.2.3

Basis (Continued).

The engineered safety features system provides a 2 of 4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a Limiting Condition for Operation (LCO) on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an AOO and factored into the derivation of the Limiting Safety System Settings and Limiting Conditions for Operation. With the rod block function out-of-service several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the Limiting Conditions for Operation (LCO) and Limiting Safety System Settings (LSSS) are valid for a Group 4 insertion of less than or equal to 15%

The operability of the Alternate Shutdown Panel (AI-185), including Wide Range Logarithmic Power and Source Range Monitors on AI-212, and Emergency Auxiliary Feedwater Panel (AI-179) instrument and control circuits ensures that sufficient capability is available to permit entry into and maintenance of the Hot Shutdown Mode from locations outside of the Control Room. This capability is required in the event that Control Room habitability is lost due to fire in the cable spreading room or Control Room.

Variances which may exist at startup between the more accurate ΔT-Power and Nuclear Instrumentation Power (NI-Power) are not significant for enabling of the trip functions. By 15% of rated power as measured by the uncalibrated NI Power, the Axial Power Distribution (APD) and Loss of Load (LOL) trip functions are enabled while the High Rate of Change of Power trip is bypassed.

The APD trip function acts to limit the axial power shape to the range assumed in the setpoint analysis. Significant margins to local power density limits exist at 15% power, as well as power levels up to at least 30% (where NI calibration occurs).

The LOL trip function acts as an anticipatory trip for the high pressurizer pressure and high power trips in order to limit the severity of a LOL transient. This trip is not credited in the USAR Chapter 14 Safety Analyses and any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

The High Rate of Change of Power trip acts to limit power excursions from low power levels and bypassing of this trip at a high power level is conservative. This trip is not credited in the USAR Chapter 14 Safety Analyses for Mode 1 operation. Any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

3.0 SURVEILLANCE REQUIREMENTS

3.1 Instrumentation and Control (Continued)

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

The minimum calibration frequencies of once-per-day (heat balance adjustment only) for the power range safety channels, and once each refueling shutdown for the process system channels, are considered adequate.

The minimum testing frequency for those instrument channels connected to the Reactor Protective System and Engineered Safety Features is based on ABB/CE probabilistic risk analyses and the accumulation of specific operating history. The quarterly frequency for the channel functional tests for these systems is based on the analyses presented in the NRC approved topical report CEN-327-A, "RPS/ESFAS Extended Test Interval Evaluation," as supplemented, and OPPD's Engineering Analysis EA-FC-93-064, "RPS/ESF Functional Test Drift Analysis."

The low temperature setpoint power operated relief valve (PORV) CHANNEL FUNCTIONAL TEST verifies operability of the actuation circuitry using the installed test switches. PORV actuation could depressurize the reactor coolant system and is not required.

Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at appropriate intervals thereafter. A single containment spray header flow rate of 1885 gpm of atomized spray is required to provide the containment response(3) specified in Section 2.4 of the Technical Specification: To achieve the 1885 gpm flow rate, no greater than ten (10) spray nozzles may be inoperable of which no more than one may be missing. Since the material is all stainless steel, normally in a dry condition, with no plugging mechanism available, retesting at appropriate intervals is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the raw water system and the containment air coolers. The SI tanks are a passive safeguard. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and are continuously monitored for satisfactory performance.

The in-containment air treatment system is designed to filter the containment building atmosphere during accident conditions. Both in-containment air treatment systems are designed to automatically start upon accident signals. Should one system fail to start, the redundant system is designed to start automatically. Each of the two systems has 100 percent capacity.(4)

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 85 percent. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR Part 100 guidelines for the accidents analyzed.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

If significant painting, fire or chemical release occurs in a ventilation zone communicating with the system that could lead to the degradation of charcoal adsorbers or HEPA filters, testing will be performed to assure system integrity and performance.