

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

- A. At least one door at each access opening is closed.
(Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment integrity.)
- B. The standby gas treatment system is operable.
- C. All automatic secondary containment isolation valves are operable or are secured in the closed position.

1.15 (DELETED)

1.16 RATED FLUX

Rated flux is the neutron flux that corresponds to a steady state power level of 1930 NW(t).
Use of the term 100 percent also refers to the 1930 thermal megawatt power level.

1.17 REACTOR THERMAL POWER-TO-WATER

Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

A. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

1.19 INSTRUMENTATION SURVEILLANCE DEFINITIONS

A. Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable.

B. Channel Test

Injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

C. Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip.

D. Source Check

A SOURCE CHECK is the qualitative assessment of channel response when the channel sensor is exposed to a source of radioactivity.

1.20 FDSAR

Oyster Creek Unit No. 1 Facility Description and Safety Analysis Report as amended by revised pages and figure changes contained in Amendments 14, 31 and 45* and continuing through Amendment 79.

1.21 CORE ALTERATION

A core alteration is the addition, removal, relocation or other manual movement of fuel or controls in the reactor core. Control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

1.22 CRITICAL POWER RATIO

The critical power ratio is the ratio of that power in a fuel assembly which is calculated, by application of an NRC approved CPR correlation, to cause some point in that assembly to experience boiling transition divided by the actual assembly operating power.

1.23 STAGGERED TEST BASIS

A Staggered Test Basis shall consist of:

- A. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

*Per Erata dtd. 4-9-69

2.3 LIMITING SAFETY SYSTEM SETTINGS

Bases:

Safety limits have been established in Specifications 2.1 and 2.2 to protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. Automatic protective devices have been provided in the plant design for corrective actions to prevent the safety limits from being exceeded in normal operation or operational transients caused by reasonably expected single operator error or equipment malfunction. This Specification establishes the trip settings for these automatic protection devices.

The Average Power Range Monitor, APRM⁽¹⁾, trip setting has been established to assure never reaching the fuel cladding integrity safety limit. The APRM system responds to changes in neutron flux. However, near the rated thermal power, the APRM is calibrated using a plant heat balance, so that the neutron flux that is sensed is read out as percent of the rated thermal power. For slow maneuvers, such as those where core thermal power, surface heat flux, and the power transferred to the water follow the neutron flux, the APRM will read reactor thermal power. For fast transients, the neutron flux will lead the power transferred from the cladding to the water due to the effect of the fuel time constant. Therefore, when the neutron flux increases to the scram setting, the percent increase in heat flux and power transferred to the water will be less than the percent increase in neutron flux.

The APRM trip setting will be varied automatically with recirculation flow, with the trip setting at the rated flow of 61.0×10^6 lb/hr of greater being 115.7% of rated neutron flux. Based on a complete evaluation of the reactor dynamic performance during normal operation as well as expected maneuvers and the various mechanical failures, it was concluded that sufficient protection is provided by the simple fixed scram setting (2,3). However, in response to expressed beliefs (4) that variation of APRM flux scram with recirculation flow is a prudent measure to ensure safe plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which could have an adverse effect on reactor safety because of the resulting thermal stresses and the unnecessary challenge to the operators. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit and yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A, when the MFLPD is greater than the fraction of the rated power (FRP). the adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

The low pressure isolation of the main steam line at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. The low-pressure isolation protection is enabled with entry into IRM range 10 or the RUN mode. In addition, a scram on 10% main steam isolation valve (MSIV) closure anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. Bypass of the MSIV closure scram function below 600 psig is permitted to provide sealing steam and allow the establishment of condenser vacuum. Advantage is taken of the MSIV scram feature to provide protection for the low-pressure portion of the fuel cladding integrity safety limit. To continue operation beyond 12% of rated power, the IRM's must be transferred into range 10. Reactor pressure must be above 825 psig to successfully transfer the IRM's into range 10. Entry into range 10 at less than 825 psig will result in main steam line isolation valve closure and MSIV closure scram. This provides automatic scram protection for the fuel cladding integrity safety limit which allows a maximum power of 25% of rated at pressures below 800 psia. Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs.

Operation of the reactor at pressure lower than 825 psig requires that the mode switch be in the STARTUP position and the IRMs be in range 9 or lower. The protection for the fuel clad integrity safety limit is provided by the IRM high neutron flux scram in each IRM range. The IRM range 9 high flux scram setting at 12% of rated power provides adequate thermal margin to the safety limit of 25% of rated power. There are few possible significant sources of rapid reactivity input to the system through IRM range 9: effects of increasing pressure at zero and low void content are minor; reactivity excursions from colder makeup water, will cause an IRM high flux trip; and the control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. In the unlikely event of a rapid or uncontrolled increase in reactivity, the IRM system would be more than adequate to ensure a scram before power could exceed the safety limit. Furthermore, a mechanical stop on the IRM range switch requires an operator to pull up on the switch handle to pass through the stop and enter range 10. This provides protection against an inadvertent entry into range 10 at low pressures. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

The adequacy of the IRM scram was determined by comparing the scram level on the IRM range 10 to the scram level on the APRMs at 30% of rated flow. The IRM scram is at 38.4% of rated power while the APRM scram is at 52.7% of rated power. The minimum flow for Oyster Creek is at 30% of rated power and this would be the lowest APRM scram point. The increased recirculation flow to 65% of flow will provide additional margin to CPR Limits. The APRM scram at 65% of rate flow is 87.1% of rated power, while the IRM range 10 scram remains at 38.4% of rated power. Therefore, transients requiring a scram based on flux excursion will be terminated sooner with a IRM range 10 scram than with an APRM scram. The transients requiring a scram by nuclear instrumentation are the loss of feedwater heating and the improper startup of an idle recirculation loop. The loss of feedwater heating transient is not affected by the range 10 IRM since the feedwater heaters will not be put into service until after the LPRM downscales have cleared, thus insuring the operability of the APRM system. This will be administratively controlled. The improper startup of an idle recirculation loop becomes less severe at lower power level and the IRM scram would be adequate to terminate the flux excursion.

The Rod Worth Minimizer is not required beyond 10% of rated power. The ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum recirculation flow of 39.65×10^6 lb/hr in range 10 a complete rod withdrawal initiated at 35% of rated power or less would not result in violating the fuel cladding safety limit. Therefore, a rod block on the IRMs at less than 35% of rated power would be adequate protection against a rod withdrawal transient.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst-case MCPR, which could occur during steady-state operation, is at 108% of the rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by (1) a reactor scram on high neutron flux, (2) failure of the recirculation pump trip on high pressure, (3) failure of the turbine bypass valves to open, and (4) failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a $\pm 1\%$ of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low level water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not operated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thus the loss of inventory is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6). The TAF definition of 353.3 inches from vessel zero is based on a fuel length of 144 inches and it is applicable to the current fuel length of 145.24 inches. The difference in fuel length does not result in changes to any analyses or set points.

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system (when the core spray system is required as identified in Section 3.4) to provide cooling water should the level drop to this point.*

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves to a load rejection and failure of the turbine bypass system. This scram is initiated by the loss of turbine acceleration relay oil pressure. The timing for this scram is almost identical to the turbine trip.

The undervoltage protection system includes a 2 out of 3 coincident logic relay designed to shift emergency buses to on-site power should normal power be degraded to an unacceptable level. There is a separate relay system designed to shift emergency buses C and D to on-site power should normal power be lost. The trip points and time delay settings have been selected to assure an adequate power source to emergency safeguards systems in the event of a total loss of normal power or degraded conditions which would adversely affect the functioning of engineered safety features connected to the plant emergency power distribution system.

The APRM downscale signal insures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associate IRM Upscale (High-High) or Inoperative signal generates a trip signal. This function is not specifically credited in the accident analyses but it is retained for overall redundancy and diversity of the RPS.

References

- (1) FDSAR, Volume 1, Section VII-4.2.4.2
- (2) FDSAR, Amendment 28, Item III.A-12

- (3) FDSAR, Amendment 32, Question 13
- (4) Letters, Peter A. Morris, Director, Division of Reaction Licensing, USAEC, to John E. Logan, Vice President, Jersey Central Power and Light Company
- (5) FDSAR, Amendment 65, Section B.XI
- (6) FDSAR, Amendment 65, Section B.IX

3.4 EMERGENCY COOLING

Applicability: Applies to the operating status of the emergency cooling systems.

Objective: To assure operability of the emergency cooling systems.

Specifications:

A. Core Spray System

1. The core spray system shall be operable at all times with irradiated fuel in the reactor vessel, except as otherwise specified in this section.
2. The absorption chamber water volume shall be at least 82,000 ft.³ in order for the core spray system to be considered operable.
3. If one core spray system loop or its core spray header delta P instrumentation becomes inoperable during the run mode, the reactor may remain in operation for a period not to exceed 7 days provided:
 - a. The remaining loop has no inoperable components and is verified daily to be operable and,
 - b. The average planar linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed 90% of the limits given in Specification 3.10.A. The action to bring the core to 90% of the APLHGR Limits must be completed within two hours after the system has been determined to be inoperable.
4. The reactor may remain in operation for a period not to exceed 15 days if one of the redundant active loop components in the core spray system becomes inoperable during the run mode provided:
 - a. In the event of an inoperable core spray booster pump, the other core spray booster pump in the loop is verified daily to be operable.
 - b. In the event of an inoperable core spray main pump, the other core spray main pump in the loop is verified daily to be operable and the APLHGR of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location shall not exceed 90% of the limits given in Specification 3.10.A. The action to bring the core to 90% of the APLHGR Limits must be completed within two hours after the component has been determined to be inoperable.
 - c. If two of the redundant active loop components become inoperable, the limits of Specification 3.4.A.3 shall apply.

5. During the period when one diesel is inoperable, the core spray equipment connected to the operable diesel shall be operable.
6. If Specifications 3.4.A.3, 3.4.A.4, and 3.4.A.5 are not met, the reactor shall be placed in the cold shutdown condition. If the core spray system becomes inoperable, the reactor shall be placed in the cold shutdown condition and no work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.
7. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is: (a) maintained in the cold shutdown condition or (b) in the refuel mode with the reactor coolant system maintained at less than 212°F and vented, and (c) no work is performed on the reactor vessel and connected systems that could result in lowering the reactor water level to less than 4'8" above the top of the active fuel. Reduced Core Spray System Availability is minimally defined as follows:
 - a. At least one core spray pump, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations.
 - b. The fire protection system is operable, and
 - c. These systems are demonstrated to be operable on a weekly basis.
8. If necessary to accomplish maintenance or modifications to the core spray systems, their power supplies or water supplies, reduced system availability is permitted when the reactor is in the refuel mode with the reactor coolant system maintained at less than 212°F or in the startup mode for the purposes of low power physics testing. Reduced core spray system availability is defined as follows:
 - a. At least one core spray pump in each loop, and system components necessary to deliver rated core spray to the reactor vessel, must remain operable to the extent that the pump and any necessary valves in each loop can be started or operated from the control room or from local control stations.
 - b. The fire protection system is operable and,
 - c. Each core spray pump and all components in 3.4.A.8a are demonstrated to be operable every 72 hours.

6. With one standby gas treatment system circuit inoperable:
 - a. During Power Operation:
 - (1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and
 - (2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.
 - (3) Restore the inoperable standby gas treatment circuit to operable status within 7 days.
 - b. During Refueling:
 - (1) Verify the operability of the other standby gas treatment system within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and
 - (2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.
 - (3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes
7. If Specifications 3.5.B.5 and 3.5.B.6 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

The technical specifications allow for torus repair work or inspections that might require draining of the suppression pool when all irradiated fuel is removed or when the potential for draining the reactor vessel has been minimized. This specification also provides assurance that the irradiated fuel has an adequate cooling water supply for normal and emergency conditions with the reactor mode switch in shutdown or refuel whenever the suppression pool is drained for inspection or repair.

The function of the primary containment isolation valves (PCIVs), in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that primary containment function assumed in the safety analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system.

The opening of locked or sealed closed containment isolation valves on an intermittent basis will be performed under administrative control including the following considerations: 1) an operator, who is in constant communication with the control room, will be stationed at the valve controls; 2) that operator will be instructed to close those valves in an accident situation; and, 3) it will be assured that environmental conditions will not preclude access to close those valves and that this action will prevent the release of radioactivity outside the containment.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber, and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the 2 psi external design pressure of the drywell; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section 11).

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety system or component be OPERABLE whenever the systems they protect are required to be OPERABLE.

The purpose of an engineering evaluation is to determine if the components protected by the snubber were adversely affected by the inoperability of the snubber. This ensures that the protected component remains capable of meeting the designed service. A documented visual inspection will usually be sufficient to determine system OPERABILITY.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements.

Secondary containment⁽⁵⁾ is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

When secondary containment is not maintained, the additional restrictions on operation and maintenance give assurance that the probability of inadvertent releases of radioactive material will be minimized. Maintenance will not be performed on systems which connect to the reactor vessel lower than the top of the active fuel unless the system is isolated by at least one locked closed isolation valve.

The trunnion room door is not an access opening for the passage of personnel and equipment into the reactor building. During all modes of operation, the trunnion room is a low traffic area and momentary openings of the door would be limited and administratively controlled and have little effect on SGTS and HVAC.

The standby gas treatment system⁽⁶⁾ filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

3.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to the operating status of the auxiliary electrical power supply.

Objective: To assure the operability of the auxiliary electrical power supply.

Specifications:

- A. The reactor shall not be made critical unless all of the following requirements are satisfied:
1. The following buses or panels energized.
 - a. 4160 volt buses 1C and 1D in the turbine building switchgear room.
 - b. 460 volt buses 1A2, 1B2, 1A21, 1B21 vital MCC 1A2 and 1B2 in the reactor building switchgear room: 1A3 and 1B3 at the intake structure; 1A21A, 1B21A, 1A21B, and 1B21B and vital MCC 1AB2 on 23'6" elevation in the reactor building; 1A24 and 1B24 at the stack.
 - c. 208/120 volt panels 3, 4, 4A, 4B, 4C and VACP-1 in the reactor building switchgear room.
 - d. 120 volt protection panel 1 and 2 in the cable room.
 - e. 125 volt DC distribution centers C and B, and panel D, Panel DC-F, isolation valve motor control center DC-1 and 125V DC motor control center DC-2.
 - f. 24 volt D.C. power panels A and B in the cable room.
 2. One 230 KV line is fully operational and switch gear and both startup transformers are energized to carry power to the station 4160 volt AC buses and carry power to or away from the plant.
 3. An additional source of power consisting of one of the following is in service connected to feed the appropriate plant 4160 V bus or buses:
 - a. A second 230 KV line fully operational.
 - b. One 34.5 KV line fully operational.
 4. Station batteries B and C and an associated battery changer are operable. Switchgear control power for 4160 volt bus 1D and 460 volt buses 1B2 and 1B3 are provided by battery B. Switchgear control power for 4160 volt bus 1C and 460 volt buses 1A2 and 1A3 are provided by battery C.
 5. Bus tie breakers ED and EC are in the open position.
- B. The reactor shall be placed in the cold shutdown condition if the availability of power falls below that required by Specification A above, except that
1. The reactor may remain in operation for a period

3.17 Control Room Heating, Ventilating, and Air-Conditioning System

Applicability: Applies to the operability of the control room heating, ventilating, and air conditioning (HVAC) system.

Objective: To assure the capability of the control room HVAC system to minimize the amount of radioactivity from entering the control room in the event of an accident.

Specifications:

- A. The control room HVAC system shall be operable during all modes of plant operation.
- B. With one control room HVAC system determined inoperable:
 - 1. Verify once per 24 hours the partial recirculation mode of operation for the operable system, or place the operable system in the partial recirculation mode; and
 - 2. Restore the inoperable system within 7 days, or prepare and submit a special report to the Commission in lieu of any other report required by Section 6.9, within the next 14 days, outlining the action taken, the cause of the inoperability and the plans/schedule for restoring the HVAC system to operable status.
- C. With both control room HVAC systems determined inoperable.
 - 1. During Power Operation: place the reactor in the cold shutdown condition with 30 hours
 - 2. During Refueling:
 - (a) Cease irradiated fuel handling operations; and
 - (b) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

Basis:

The operability of the control room HVAC system ensures that the control room will remain habitable for operations personnel during a postulated design basis accident. The control room envelope includes the control room panel area, the shift supervisor's office, toilet room, kitchen, and lower cable spreading room. Since Systems A and B do not have HEPA filters or charcoal absorbers, the supply fan and dampers for each system minimize the beta and gamma doses to the operators by providing positive pressurization and limiting the makeup and infiltration air into the control room envelope. For the supply of 100% outside air to the control room envelope, the dose increase to 29.1 rem beta and 3.14 rem gamma for the assumed 30 days; however, these values are within the allowable limits.

of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening- closing cycle, the position indicating lights are designed to function as follows:

Full Closed (Closed to 0.10" open)	2 Green - On 2 Red - Off
Open 0.10" (0.10" open to full open)	2 Green - Off 2 Red - On

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about 1/10th of the design lifetime is extremely conservative. The alarm systems for the vacuum breakers will be calibrated during each refueling outage. This frequency is based on experience and engineering judgement.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain a 1/4 inch of water vacuum, tests the operation of the reactor building isolation valves, leakage tightness of the reactor building and performance of the standby gas treatment system. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing the reactor building in leakage test prior to refueling demonstrates secondary containment capability prior to extensive fuel handling operations associated with the outage. Verifying the efficiency and operation of charcoal filters once per 18 months gives sufficient confidence of standby gas treatment system performance capability. A charcoal filter efficiency of 99% for halogen removal is adequate.

The in-place testing of charcoal filters is performed using halogenated hydrocarbon refrigerant which is injected into the system upstream of the charcoal filters. Measurement of the refrigerant concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indication of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Cleaning Conference.*

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential releases of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP at testing medium.

*D.R. Muhàbier, "In Place Nondestructive Leak Test for Iodine Adsorbers," Proceedings of the Ninth AEC Air Cleaning Conference, USAEC Report CONF-660904, 1966