Docket Files

APR 1 9 1974

Docket Nos. 50-237 and 50-249

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Nuclear Licensing Administrator -
Boiling Water Reactors
Post Office Box 767
Chicago, Illinois 60690

Change No. 28
License No. DPR-19
Change No. 19
License No. DPR-25

Gentlemen:

By letters dated February 22, 1972, and February 20, 1973, relative to Dresden Units 2 and 3, you requested approval of proposed containment atmospheric dilution (CAD) systems and related technical specifications. By supplements dated April 30, August 20, and August 29, 1973, you provided additional information at our request. We have reviewed your application and hereby authorize installation and operation of the CAD systems. We are deferring revision of technical specification requirements on CAD system operability until the system is installed and tested. In the interim, we are placing requirements on the existing nitrogen addition, purge and monitoring systems.

By applications dated September 20 and 21, 1973, you requested proposed changes to the Technical Specifications for Dresden Units 2 and 3 (respectively) regarding limiting conditions of operation and surveillance requirements for primary containment vacuum breakers. Relevant information was also provided in Special Reports No. 23 dated April 23, 1973, and No. 23A dated August 1, 1973. We have evaluated your proposals and made several modifications. Based on our evaluation, we have concluded that the proposed specifications, as modified, provide acceptable assurance that the containment vacuum breakers will function as required.

As a result of our continuing review of technical specifications we also have revised requirements regarding main steam line tunnel radiation monitors, the standby liquid control system, the core and containment cooling systems, reactor coolant chemistry, jet pumps and reporting requirements. These changes are consistent with current requirements at other boiling water reactors and, in our opinion as reflected in our evaluation, enhance safety.

The staff evaluation of the acceptability of the CAD system and the revision to the technical specifications are enclosed.

Based on the enclosed evaluation, we have concluded that operation with the proposed CAD system and in accordance with the revised technical specifications attached hereto, does not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered.

Pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License Nos. DPR-19 and DPR-25 are hereby changed as follows:

1. License No. DPR-19 - Replace pages 38, 39, 47, 66-68, 71, 77, 79, 80, 81, 85, 89, 91, 91a, 99, 99a, 116, 117, 126, 127, 131, 177-185 with the enclosed revised pages 38, 39, 47, 66-68, 71, 77, 79, 80, 81, 85, 89, 91, 91a, 99, 99a, 116, 116a, 117, 117a, 117b, 126, 126a, 127, 131, 131a, 131b, 177 and 178.

2. License No. DPR-25 - Replace pages 38, 39, 47, 66-68, 71, 77, 79, 80, 81, 85, 89, 91, 99, 116, 117, 126, 127, 131, 177-185 with the enclosed revised pages 38, 39, 47, 66-68, 71, 77, 79, 80, 81, 85, 89, 91, 91a, 99, 99a, 116, 116a, 117, 117a, 117b, 126, 126a, 127, 131, 131a, 131b, 177 and 178.

We request that within 60 days of the date of this change you provide your schedule for installation and testing of the CAD system.

Sincerely,

/S/

Karl R. Goller
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages as stated above

cc w/enclosures: See next page

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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NOS. 50-237 AND 50-249

COMMONWEALTH EDISON COMPANY

INTRODUCTION

As a result of continued evaluation of the Dresden Units 2 and 3, the staff has initiated several changes to the Specifications of License Nos. DPR-19 and DPR-25. The Specifications involve requirements on main steam line tunnel radiation monitors, standby liquid control systems, core and containment cooling systems, reactor coolant water chemistry, jet pumps, containment vacuum breakers, post loss of coolant accident containment atmospheric control, and reporting requirements. Each of these changes is discussed below:

Section 3.2

The limiting scram trip level setting for the main steam line tunnel radiation monitor is being reduced from 7 times to 3 times background at normal rated power. Also, a requirement for a limiting alarm trip setting at 1.5 times background is being added. Our evaluation indicates that these settings will provide early warning and increased plant protection in the event of a release of radioactive material without an undue increase of reactor scrams. The setting will be consistent with those required at other boiling water reactors.

Sections 3.4 and 4.4

The limiting conditions of operation and surveillance requirements for the standby liquid control system are being revised to conform with requirements for other boiling water reactors. Section 3.4.B is being revised to eliminate a reporting requirement as an alternate to shutdown if certain operability requirements are not met. Section 3.4.D is being revised to specify a time limit within which shutdown must occur if operability requirements are not met. Section 4.4.A is being revised to require more complete testing of the standby liquid control system including in-place test firing of the explosive actuated valves.

Sections 3.5.B, C, D, E, F, and G

The limiting conditions of operation for core and containment cooling systems now include a reporting requirement as an alternative to shutdown if certain operability requirements are not met within a specified period of time. The reporting alternative is being removed so that the specified systems must either be operable or the unit brought to a shutdown condition. This conforms with present requirements at other power reactors.

Section 3.6.C.4

The reactor coolant conductivity and chloride ion limits with steaming rates greater than or equal to 100,000 pounds per hour are being halved to conform with Regulatory Guide 1.56. The basis for these limitations is discussed in Regulatory Guide 1.56.

Sections 3.6.G and H

The specifications related to jet pump operability and surveillance are being revised to provide greater assurance that the reactor is not being operated with a failed jet pump. Shutdown and repair in the event of a failed jet pump is necessary because some jet pump failures can increase the cross sectional area for blowdown following the design basis loss of coolant accident. The surveillance requirements when operating with one recirculation pump with the equalizer valves closed are being clarified.

Sections 3.7.A and 4.7.A

By applications dated September 20 and 21, 1973, Commonwealth Edison (CE) requested proposed changes to the Technical Specifications for Dresden Units 2 and 3, respectively, regarding limiting conditions of operation and surveillance requirements for primary containment vacuum breakers. The requested changes were in response to a review initiated by our letter to CE dated January 12, 1973. A special report and supplement submitted by CE's letters of April 23 and August 1, 1973, in response to our letters of January 12 and June 6, 1973, provide additional vacuum breaker information which we have evaluated.

The drywell vacuum relief valves (vacuum breakers) are installed to assure that the pressure in the torus suppression chamber does not exceed the pressure in the drywell sections of the primary containment by more than about 0.5 psid. In the event of an accident causing a pressure increase in the drywell, the vacuum breakers should be closed so that condensible gases in the drywell pass into the suppression pool. Bypassing of the condensible gases through the vacuum breakers should be limited to limit containment pressure. Analyses submitted by CE's letter of April 23, 1973, indicated that the maximum allowable bypass area between the drywell and torus is 0.3 square foot. We concur that this maximum bypass area is acceptable.

Commonwealth has informed us that vacuum breaker position limit switches with alarms have been installed in the control room to allow prompt detection and corrective action if a vacuum breaker opening is excessive. In addition, they have proposed limiting conditions of operation and surveillance requirements for both opening and closure. We have reviewed their proposed specifications and made several modifications. The limiting conditions for operation of the pressure suppression chamber-drywell vacuum breakers will include new requirements for valve closure and valve position alarms. Surveillance requirements have been added for valve operability, valve position indicator operability, vacuum breaker inspection and leak rate tests.

We have also reviewed the specifications for the pressure suppression chamber-reactor building vacuum breakers. As a result of this review, we have added specifications related to opening force and have added surveillance requirements including inspection and tests of force required for opening.

Containment Atmospheric Dilution System and Sections 3.7.A.5, 3.7.A.6 and 4.7.A.6

By letters dated February 22, 1972, and February 20, 1973, relative to Dresden Units 2 and 3, Commonwealth Edison requested approval of proposed containment atmospheric dilution (CAD) systems and technical specifications to limit combustible gas concentrations in containment in the event of a loss-of-coolant accident. By supplements dated April 30, August 20, and August 29, 1973, CE provided additional information at our request. The need for a study of a system for combustible gas control was identified by the staff in its Safety Evaluation for the Dresden 3 Operating License.

We have reviewed the application with supplements and have concluded that the proposed CAD and monitoring systems are acceptable with respect to reliability criteria for emergency safeguard systems. We also have concluded that the systems provide acceptable assurance that combustible gas concentrations following a loss-of-coolant accident can be maintained below the guideline limits set forth in Regulatory Guide 1.7 provided initial oxygen concentrations in containment is limited to 4%. The 4% limit is based on our independent analyses and is consistent with limitations found necessary at similar reactors.

We have performed independent dose analysis of possible accident doses due to containment purging with the CAD system operational. With initial oxygen concentration at 4% and with the conservative assumption that containment leak rate is zero, we have calculated that purge initiation would be required 18 days after the postulated loss-of-coolant accident at a rate of 9787 standard cubic feet per day to maintain the containment pressure less than the repressurization limit of 26 psig. Assuming a continuous containment purge for 30 days following initiation of the above purge rate, we calculated a purge dose contribution at the Low Population Zone distance from Dresden to be less than five Rem-thyroid and under one Rem whole body. The sum of these doses and the previously calculated doses from the postulated loss-of-coolant accidents at Dresden is well under Part 100 guidelines and is acceptable.

Based on our evaluation, we are modifying the limiting conditions for operation for oxygen concentration (3.7.A.5) and adding Technical Specification requirements for the existing containment nitrogen makeup and purge systems (3.7.A.6 and 4.7.A.6). The existing systems are not designed as engineered safety systems but provide an additional measure of protection until the CAD system is installed. After installation of the CAD system the Technical Specification requirements will be revised to apply to this system.

We have concluded that the containment atmospheric dilution systems, installed and operated as proposed but with an initial containment oxygen concentration of 4%, provide acceptable assurance that combustible gas concentrations and purge doses will be below guideline limits following a loss-of-coolant accident.

Section 6.6

Section 6.6, Reporting Requirements, is being revised in its entirety to conform with the reporting requirements of Regulatory Guide 1.21, 4.1 and with one exception, 1.16. Submission of FSAR changes as stated in Regulatory Guide 1.16 is not being required at this time pending further review of our requirements. In addition to the reports required by the Regulatory Guides, requirements for special reports presently in the specifications will be retained.

CONCLUSION

We have concluded that the addition of the Containment Atmospheric Dilution System and the revisions of the technical specifications discussed above will enhance the safety of operation. Based on the evaluations above, we also have concluded that operation of the reactors with the CAD systems and in accordance with the revised Technical Specifications discussed herein, does not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered.

Richard D. Silver
Operating Reactors Branch #2
Directorate of Licensing

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: April 19, 1974

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum No. of Operable Inst. Channels per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	>143" above top of active fuel	A
2	Reactor Low Low Water	≥83" above top of active fuel	A
2	High drywell pressure	≤2 psig rated (4), (5)	A
2 (2)	High Flow Main Steam line	≤120% of rated steam flow	B
2 of 4 in each of 4 sets	High Temperature Main Steam Line Tunnel	≤200°F	B
2	High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated power background	B
2	Low Pressure Main Steamline	≥850 psig	B
	High Flow Isolation Condenser Line		
1	Steamline Side	≤20 psi diff. on steamline side	C
1	Condensate Return Side	≤ 32" water diff. on condensate return side	C
2	High Flow HPCI Steam Line	≤150" water	D
4	High Temperature HPCI Steam Line Area	≤200°F	D

Notes:

1. Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the RUN position.
2. Per each steamline.
3. Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped.

TABLE 3.2.1 (cont)

If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

A. Initiate an orderly shutdown and have reactor in cold shutdown condition in 24 hours.

B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.

C. Close isolation valves in isolation condenser system.

D. Close isolation valves in HPCI subsystem.

4. Need not be operable when primary containment integrity is not required.

5. May be bypassed when necessary during purging for containment inerting and deinerting.

6. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

Section 6.2.7.1 and 14.2.4.2 SAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Ref. Section 6.2.7.1 SAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of Group I primary system isolation valves.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Ref. Sections 14.2.3.9 and 14.2.3.10 SAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Ref. Section 14.2.1.7 SAR. The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Atomic Energy Commission.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 850 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Ref. Section 11.2.3 SAR.

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

During periods when fuel is in the reactor the standby liquid control system shall be operable except when the reactor is in the Cold Shutdown Condition and all control rods are fully inserted and Specification 3.3.A is met or as specified in 3.4.B below.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

Specification:

A. Normal Operation

The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per month-

Demineralized water shall be recycled to the test tank. Pump minimum flow rate of 39 gpm shall be verified against a system head of 1275 psig.

3.4 LIMITING CONDITION FOR OPERATION

4.4 SURVEILLANCE REQUIREMENT

2. At least once during each operating cycle

a. Actuate one of the two standby liquid control systems using the normal actuation switch and pump demineralized water into the reactor vessel. Pump minimum flow rate shall be verified against a previous test at the same reactor vessel pressure. The replacement charges will be selected from a batch from which at least one charge has been successfully test fired and which will not exceed five years life when their use is terminated. Both systems shall be tested and inspected, including each explosive actuated valve, in the course of two operating cycles.

b. Test that the setting of the system pressure relief valves is between 1400 and 1490 psig.

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted provided that the component is returned to an operable condition within 7 days.

B. Surveillance with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

3.4 LIMITING CONDITION FOR OPERATION

C. The liquid poison tank shall contain a boron bearing solution that satisfies the volume-concentration requirements of Figure 3.4.1 and at all times when the standby liquid control system is required to be operable and the solution temperature including that in the pump suction piping shall not be less than the temperature presented in Figure 3.4.2.

D. If specification 3.4.A through C are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.4 SURVEILLANCE REQUIREMENT

C. The availability of the proper boron bearing solution shall be verified by performance of the following tests:

1. At least once per month — Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

2. At least once per day — Solution volume shall be checked.

3. At least once per day — The solution temperature shall be checked.

Bases:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of 720 ppm of boron in the reactor core in less than 100 minutes. 720 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 3%Δk subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional margin (25%) for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3478 gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (100 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required pumping rate of 39 gallons per minute, the maximum storage volume of the boron solution is established as 4,059 gallons (158 gallons are contained below the pump suction and, therefore, cannot be inserted).

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges have not deteriorated, the actuation circuit is functioning properly, the valve functions properly, and no flow blockages exist. The replacement charge will be selected from a batch for which there has been a successful test firing. Recommendations of the vendor shall be followed in maintaining a five-year life of the explosive charges. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psig protection from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily.

C. The solution saturation temperature of 13% sodium pentaborate, by weight, is 59°F. To guard against boron precipitation, the solution including that in the pump suction piping is kept at least 10°F above the saturation temperature by a tank heater and by heat tracing in the pump suction piping. The 10°F margin is containment cooling subsystem, both core spray subsystems and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of 3.5.B cannot be met an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

C. HPCI Subsystem

1. Except as specified in 3.5.C.2 below, the HPCI subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that the HPCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Pressure Relief Subsystem, the core spray subsystems, LPCI subsystem, and isolation cooling system are operable.

3. If the requirements of 3.5.C cannot be met an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

C. Surveillance of HPCI Subsystem shall be performed as follows:

1. HPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that the HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig to 150 psig.

2. When it is determined that HPCI subsystem is inoperable, the LPCI subsystem, both core spray subsystems, the automatic pressure relief subsystem, and the motor operated isolation valves and shell side make-up system for the isolation condenser system shall be demonstrated to be operable immediately. The automatic pressure relief and motor operated isolation valves and shell side make-up system of the isolation condenser shall be demonstrated to be operable daily thereafter.

3.5 LIMITING CONDITION FOR OPERATION

4. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

E. Isolation Condenser System

1. Whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.F.2.

2. From and after the date that the isolation condenser system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the HPCI subsystem are operable.

3. If the requirements of 3.5.E cannot be met an orderly shutdown shall be

4.5 SURVEILLANCE REQUIREMENT

E. Surveillance of the Isolation Condenser System shall be performed as follows:

1. Isolation Condenser System Testing:

   a. The shell side water level and temperature shall be checked daily.

   b. Simulated automatic actuation and functional system testing shall be performed during each refueling outage or whenever major repairs are completed on the system.

   c. The system heat removal capability shall be determined once every five years.

   d. Calibrate vent line radiation monitors quarterly.

2. When it is determined that the isolation condenser system is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately and daily thereafter.

initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

F. Minimum Core and Containment Cooling System Availability

1. During any period when the unit or shared diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days provided that all of the low pressure core cooling and containment cooling subsystems shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

2. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and reactor is in the cold shutdown condition, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

4. When irradiated fuel is in the reactor vessel and the reactor is in the refuel condition, the torus may be drained completely

F. Surveillance of Core and Containment Cooling System

1. When it is determined that either the unit or shared diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

2. Actions necessary to assure that the plant can be safely shut down and maintained in this condition in case of failure of the Dresden Dam shall be demonstrated to be adequate every third refueling outage. If this Specification has been complied with for Dresden Unit 3, it shall not be required for Dresden Unit 2.

and control rod drive maintenance performed provided that the spent fuel pool gates are open, the fuel pool water level is maintained above the low level alarm point, and the minimum total condensate storage reserve is maintained at 230,000 gallons, and provided that not more than one control rod drive housing is open at one time, the control rod drive housing is blanked following removal of the control rod drive, no work is being performed in the reactor vessel while the housing is open and a special flange is available which can be used to blank an open housing in the event of a leak.

G. (Deleted)

H. Maintenance of Filled Discharge Pipe

Whenever core spray, LPCI, or HPCI ECCS are required to be operable, the discharge piping from the pump discharge of these systems to the last check valve shall be filled.

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray, LPCI, and HPCI are filled:

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

F. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

G. (Deleted)

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.3:

Conductivity	2 μmho/cm
Chloride ion	0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24 hours after placing the reactor in the power operating condition.

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour.

Conductivity	5 μmho/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met,

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

D. Coolant Leakage

Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.

3.6 LIMITING CONDITION FOR OPERATION

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

F. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3.6 LIMITING CONDITION FOR OPERATION

H. Recirculation Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

Recirculation pumps speed shall be checked daily for mismatch.

These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the AEC.

G. <u>Jet Pumps</u> - Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

H. Jet Pump Flow Mismatch

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

In addition, during the start-up of Dresden Unit 2 it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Except as specified in Specifications 3.7.A.3.b below, two pressure suppression chamber - reactor building vacuum breakers in each line shall be operable at all times when the primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber - reactor building air operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

   b. From and after the date that one of the pressure suppression chamber - reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the procedure does not violate primary containment integrity.

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. The pressure suppression chamber - reactor building vacuum breakers and associated instrumentation, including setpoint, shall be checked for proper operation every three months.

   b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.3.a. and each vacuum breaker shall be inspected and verified to meet design requirements.

3.7 LIMITING CONDITION FOR OPERATION

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all pressure suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.4.b, c and d., below, pressure suppression chamber - drywell vacuum breakers shall be considered operable if:

      (1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding the equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

      (2) The valve can be closed by gravity when released after being opened by manual means, to within the equivalent of 1/16" at all points along the seal surface of the disk.

      (3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 1/16" at all points along the seal surface of the disk.

4.7 SURVEILLANCE REQUIREMENT

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic Operability Tests

      Once each month each pressure suppression chamber - drywell vacuum breaker shall be exercised. Operability of position switches, and position indicators and alarms shall be verified.

   b. During each refueling outage:

      (1) The pressure suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

      (2) Vacuum breakers position indication and alarm systems shall be calibrated and functionally tested.

      (3) At least 25% of the vacuum breakers shall be inspected such that all vacuum breakers shall have been inspected following every fourth refueling outage. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected.

b. Reactor operation may continue provided that no more than one quarter of the number of pressure suppression chamber - drywell vacuum breakers are determined to be inoperable provided that they are secured or known to be in the closed position.

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each operable vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor cooling pressure above 90 psig, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run Mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

(4) A drywell to suppresion chamber leak test shall demonstrate that with initial differential pressure of not less than 1.0 psi, the differential pressure decay rate does not exceed the rate which would occur through a 1-inch orifice without the addition of air or nitrogen.

5. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITION FOR OPERATION

6. Containment Atmospheric Dilution and Purge

a. Whenever the reactor is in power operation the normal containment makeup inerting system shall be operable and capable of supplying nitrogen to containment for atmosphere dilution if required by post LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

b. Whenever either Unit 2 or 3 is in power operation, the containment makeup inerting system nitrogen storage tank level liquid level shall be equal to or greater than 60 inches. If this minimum level cannot be met, the minimum level shall be restored within 7 days or both Unit 2 and Unit 3 shall be taken out of power operation. During such seven day interval the minimum level shall be 20 inches or both Unit 2 and 3 shall be taken out of power operation.

c. Whenever the reactor is in power operation, the primary containment purge system shall be operable. If this specification cannot be met the reactor must be taken out of power operation.

4.7 SURVEILLANCE REQUIREMENT

6. Containment Atmospheric Dilution and Purge

a. Once a month, the valves in the nitrogen makeup system shall be actuated to determine operability.

b. The level in the liquid N_2 storage tank shall be recorded weekly and after reinerting containment.

c. Once a month, the valves in the purge line to the standby gas treatment system shall be actuated to determine operability.

d. Whenever the reactor is in power operation, the primary containment oxygen sampling system shall be operable. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

e. The maximum containment repressurization pressure using the containment makeup inerting system shall be 26 psig.

7. If the specifications of 3.7.A cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

d. The containment oxygen analyzing system shall be functionally tested once per week and shall be calibrated once per 6 months.

Bodega Bay tests were 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for emergency core cooling systems operability as explained in basis 3.5.F.

Using a 50°F rise (Section 5.2.3.1 SAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F a temperature of 145°F is achieved which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps) containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps.

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperatures above 212°F provides additional margin above that available at 330°F.

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 structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% flow pipes each containing two vacuum relief breakers. Operation of either flow pipe will maintain the pressure differential less than 1 psig, the external design pressure of the primary containment. Redundancy of lines justifies reactor operation with one valve out of service for repairs for a period of seven days.

The capacity of the pressure suppression chamber - drywell vacuum breakers is designed to limit the pressure differential between the suppression chamber and drywell to not greater than 0.5 psi during post-accident drywell cooling. They are sized on the basis of the Bodega Bay pressure suppression system test.

Based on these tests, design flow from the suppression chamber to the drywell can be obtained with three (3) of the vacuum breakers closed without exceeding the 0.5 psi differential pressure limit.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable a
The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16" at all points along the seal surface of the disk (see Dresden Special Report No. 23).

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE 279 standards. The quality of the alarm system justifies continued reactor operation for 15 days between differential pressure decay rate tests if one alarm system is inoperable for one or more operable vacuum breakers.

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss of coolant accident would lead to the liberation of sufficient hydrogen to a result in a flammable concentration in the containment. Subsequent ignition of the hydrogen if it is present in sufficient quantities to result in excessively rapid recombination, could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture remains below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. During an interim period prior to installation of the Containment Atmospheric Dilution (CAD) system the normal inerting nitrogen makeup system will be available for post-LOCA nitrogen injection.

By maintaining a minimum level of 60 inches in the liquid nitrogen storage tank, a minimum of 200,000 cubic feet of nitrogen is assured which corresponds to a seven day supply. During reinerting of containment the supply may temporarily drop below a seven day supply but at no time is the inventory to drop below a minimum of a two day supply (20 inch level). By normally maintaining at least a 7-day supply of nitrogen on site and maintaining a minimum of a two day supply there will be assurance of sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply.

A system for controlled purging through the Standby Gas Treatment system is necessary to limit repressurization pressure from post LOCA nitrogen addition in a manner which will limit offsite doses. Controlled purging also provides a backup method of controlling hydrogen concentration.

A means to determine post LOCA containment oxygen concentration is necessary to readily enable the reactor operator to take appropriate action to control containment atmosphere. In the interim, prior to installation of the CAD and associated monitoring systems, the containment oxygen analyzing system will be available.

The maximum containment repressurization pressure of 26 psi provides adequate margin to containment design pressure and a delay time prior to purge which results in acceptable purge doses.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and O_2 concentration exceeds 4%.

B. Standby Gas Treatment System and C Secondary Containment - The secondary containment 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; the reactor building 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 complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The standby gas treatment system is designed to filter and exhaust 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. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 1/4-inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 200% capacity. Ref. Section 5.3.2 SAR. If one standby gas treatment system circuit is inoperable, the other circuit will be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the plant is brought to a condition where the system is not required.

While only a small amount of particulates are released from the pressure suppression chamber system as a result of the loss of coolant

SAR indicates that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leak in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the reactor building - pressure suppression chamber vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness test). These vacuum breakers are normally in the closed position and open only during tests or a post accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgment.

Pressure suppression chamber - drywell vacuum breakers monthly operability tests are performed to check capability of the disks to open and close and to verify that the position indication and alarm circuits function properly. The disk opens during post accident conditions and occasionally during transient additions of energy to the torus through relief valves. This infrequent operation of the disks and the quality of equipment justify the frequency of operability tests of this equipment.

Measurement of force to open, calibration of position switches, inspection of equipment and functional testing are performed during each refueling outage. This frequency is based on equipment quality, experience and judgment. Also a stringent differential pressure decay rate test is performed during refueling outages. This test is performed to verify that total leakage paths between the drywell and suppression chamber are not in excess of the equivalent to a 1-inch orifice.

This small leakage path is only a small fraction of the allowable, thus integrity of the containment system is assured prior to startup following each refueling outage (See Dresden Special Report No. 23).

When a suppression chamber-drywell vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panel are designed to function as follows:

Full Closed (Closed to $\leq$1/16" open)	2 Green - On
Intermediate Position (>1/16" open to full open)	2 Green - Off

The remote test panel consists of two green lights for each of the twelve valves. The two switches controlling the green lights are adjusted to provide indication and alarm if a disk opening occurs that is equivalent to one-sixteenth of an inch (1/16")at all points around the circumference of the valve disk. The control room alarm circuits for each vacuum breaker are redundant and fail safe. This assures that no single failure will defeat alarming the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or that system degradation has occurred and additional testing is required immediately. The frequency of testing the alarms is based on experience and quality of the equipment. During each refueling outage, three drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in 1/10 of the design lifetime is extremely conservative.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

Recording N_2 storage tank level weekly and after containment reinerting provides assurance of an adequate onsite supply.

Weekly testing of the oxygen analyzer and monthly actuation of the nitrogen makeup and purge line valves provides assurance of operational readiness.

B. Standby Gas Treatment System and
C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus, reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after 2 years of operation in the rugged shipboard environment of the NS Savannah (ORNL 3726). Pressure drop tests across filter sections are performed to detect gross plugging or leak paths through the filter media. Considering the relatively short time that the fans may be run for test purposes, plugging is unlikely, and the test interval of once per operating cycle is reasonable. Duct heater tests will be conducted once during each operating cycle. Considering the simplicity of the heating circuit, the test frequency is sufficient.

The in-place testing of charcoal filters is performed using Freon-112 or equivalent, which is injected into the system upstream of the charcoal filters. Measurements of the Freon concentration upstream and downstream of the charcoal filters is made. 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 iodine, the test also gives an indication of the relative efficiency of the installed system.

High-efficiency particulate filters are installed before and after the charcoal filters to minimize potential release 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

6.6 PLANT REPORTING REQUIREMENTS

A. The reporting information to be submitted to the USAEC in addition to the reports required by Title 10, Code of Federal Regulations shall be in accordance with the Regulatory Positions of Regulatory Guide 1.16, "Reporting of Operating Information", Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants", and Regulatory Guide 4.1, "Measuring and Reporting of Radioactivity in the Environs of Nuclear Power Plants". Changes in the reporting requirements which occur as a result of revisions to the above Regulatory Guides shall be incorporated into the reporting information no later than one reporting period following the reporting period in which the revision is issued. Submission of FSAR changes as stated in Regulatory Guide 1.16 shall not be required.

B. Special reports shall be submitted in writing within 90 days to the Directorate of Licensing, USAEC, Washington, D. C. 20545 as indicated in Table 6.6.1.

6.0 ADMINISTRATIVE CONTROLS

TABLE 6.6.1
SPECIAL REPORTS

AREA	SPECIFICATION REFERENCE	SUBMITTAL DATE
a. Response time of safety related instruments(2)	1.0.E (Dresden 1)	Semiannual report.
b. Main steam isolation valve and feedwater power operated isolation valves closure times(2)	3.7.B.1.c (Dresden 1)	Semiannual report.
c. Primary coolant leakage to drywell(4)	4.6.D Bases	5 years (1)
d. Inservice inspection evaluation(4)	Table 4.6.1	5 years (1)
e. Evaluation of EGCS operation(4)	3.3.F Bases	Upon completion of initial testing.
f. Failed fuel detection(4)	3.2 Bases	5 years (1)
g. Main steam line leakage to steam tunnel(4)	4.6.D Bases	5 years (1)
h. Inservice inspection development(4)	4.6.1 Bases	5 years (1)
i. Inservice inspection of sensitized stainless steel components(3)	4.6.F	4 years (1)

NOTES:

(1) The report shall be submitted within the period of time listed based on the commercial service date as the starting point.
(2) Dresden 1 only.
(3) Dresden 2 only.
(4) Dresden 2 and 3 only.