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Amdt. 102 to DPR-25

August 10, 1989

Docket Nos. 50-237
and 50-249

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Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENTS RELATED TO MULTIPLE TESTING REQUIREMENTS OF THE ECCS AND SGTS - DSIP 2 (TAC NOS. 71610 AND 71611)

Re: Dresden Nuclear Power Station, Unit Nos. 2 and 3

The Commission has issued the enclosed Amendment No. to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. to Facility Operating License No. DPR-25 for Dresden Unit 3. The amendments are in response to your application dated December 21, 1988 as supplemented by your May 4, 1989 submittal.

The amendments primarily revise the testing requirements for other systems or subsystems of the Emergency Core Cooling System (ECCS) or Standby Gas Treatment System (SGTS) when one system or subsystem is inoperable. In addition, operability requirements for several ECCS systems or subsystems were revised and some administrative changes were made.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Byron Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
1. Amendment No. 107 to License No. DPR-19
2. Amendment No. 102 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

PDIII-2:LA	PDIII-2:PM	PDIII-2:(A)PD	OGC
LLuther:dm	BSiegel	PShemanski	
7/5/89	7/6/89	7/6/89	7/11/89

Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

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COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated December 21, 1988 and supplemental information provided dated May 4, 1989 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 10, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iii	iii
iv	iv
vii	vii
1.0-4	1.0-4
3/4.5-2	3/4.5-2
3/4.5-3	3/4.5-3
3/4.5-4	3/4.5-4
	3/4.5-4a
3/4.5-5	3/4.5-5
3/4.5-6	3/4.5-6
3/4.5-7	3/4.5-7
	3/4.5-7a
3/4.5-8	3/4.5-8
3/4.5-9	3/4.5-9
3/4.5-10	3/4.5-10
3/4.5-11	3/4.5-11
3/4.5-12	3/4.5-12
3/4.5-12a	3/4.5-12a
3/4.5-13	3/4.5-13
B 3/4.5-30	B 3/4.5-30
B 3/4.5-31	B 3/4.5-31
	B 3/4.5-32
B 3/4.5-32	B 3/4.5-33
B 3/4.5-33	B 3/4.5-34
B 3/4.5-34	B 3/4.5-35
B 3/4.5-35	B 3/4.5-36
B 3/4.5-36	B 3/4.5-37
B 3/4.5-37	B 3/4.5-38
B 3/4.5-38	B 3/4.5-39
B 3/4.5-39	B 3/4.5-40
B 3/4.5-40	B 3/4.5-41
B 3/4.5-41	B 3/4.5-42
B 3/4.5-42	B 3/4.5-43
B 3/4.5-43	B 3/4.5-44

AMENDMENT NO. 102 CONT'D

REMOVE	INSERT
3/4.7-19	3/4.7-19
3/4.7-20	3/4.7-20
3/4.7-22	3/4.7-22
3/4.7-23	3/4.7-23
3/4.7-24	3/4.7-24
3/4.7-25	3/4.7-25
3/4.7-26	3/4.7-26
3/4.7-27	3/4.7-27
B 3/4.7-38	B 3/4.7-38
B 3/4.7-39	B 3/4.7-39
B 3/4.7-40	B 3/4.7-40
B 3/4.7-45	B 3/4.7-45
B 3/4.7-46	B 3/4.7-46
3/4.9-3	3/4.9-3
3/4.9-4	3/4.9-4
3/4.9-5	3/4.9-5
3/4.9-6	3/4.9-6
	3/4.9-6a

(Table of Contents, Cont'd.)

(Table of Contents, Cont'd.)

List of Tables

1.0 DEFINITIONS (Cont'd.)

V. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condensor low vacuum and main steamline isolation valve closure, are by-passed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron-monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor protection system is energized with APRM protection and RBM interlocks in service.

W. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

X. Refueling-Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Y. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

Z. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is in compliance with the provisions of Specification 3.7.B.

3. All automatic ventilation system isolation valves are operable or are secured in the isolated position.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the date that one of the core spray subsystems 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 other core spray subsystem and the LPCI subsystem and the diesel generators required for operation

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

	to a reactor vessel pressure of 90 psig	
c.	Pump Operability	Once/month
d.	Motor Operated Valve	Once/month
e.	Core Spray header delta p instrumentation:	
	check	Once/day
	calibrate	Once/3 months
	test	Once/3 months
f.	Logic System Functional Test	Each Refueling Outage

2. Deleted

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

of such components if no external source of power were available shall be operable.

3. Except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.3 below, the LPCI subsystem shall be operable whenever irradiated fuel is in the reactor vessel.

4. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray subsystems and the diesel generators required for operation of such components if no external source of power were available shall be operable.

5. From and after the date that the LPCI subsystem is made or found to be inoperable

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. LPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b, c, d, and f, except that three LPCI pumps shall deliver at least 14,500 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

4. Deleted

5. Deleted

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of both core spray subsystems, the containment cooling subsystem (including 2 LPCI pumps) and the diesel generators required for operation of such components if no external source of power were available shall be operable.

6. Containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a maximum of one drywell spray loop may be inoperable for thirty days when the reactor water temperature is greater than 212°F.

7. The doors of the core spray and LPCI pump compartments shall be closed at all times except during passage in order to consider the core spray and the LPCI subsystems operable.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

6. During each five year period an air test shall be performed on the drywell spray headers and nozzles.

7. Whenever the LPCI and core spray subsystems are required to be operable, the doors of the core spray and LPCI pump compartments shall be verified to be closed weekly.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

8. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Subsequently, the reactor mode switch may be placed in Refuel in accordance with 3.5.F.3 through 3.5.F.6.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Containment Cooling Subsystem

1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 through 3.5.F.6 below, both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the containment cooling service water subsystem pumps is made or found to be inoperable for

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Surveillance of the Containment Cooling Subsystem shall be performed as follows:

1. Containment Cooling Service Water Subsystem Testing:

	Item	Frequency
a.	Pump & Valve Operability	Once/3 months
b.	Flow Rate Test. Each containment cooling water pump shall deliver at least 3500 gpm against a pressure of 180 psig.	After pump maintenance and every 3 months
c.	Each manual, power operated or automatic valve, in the flow path that is not locked, sealed or otherwise secured in its position, must be verified to be in its correct position.	Every 31 days

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the containment cooling subsystem are operable.

3. From and after the date that one containment cooling 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 all active components of the other 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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

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 150 psig and irradiated fuel is in the reactor vessel.

2. a. 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.

   b. During reactor startup when HPCI Surveillances are being performed, if the testing requirements of 4.5.C.3 or 4.5.C.4 cannot be met, continued reactor startup is not permitted. The HPCI system shall be declared inoperable and the provisions of 3.5.C.3 shall be implemented.

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

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Surveillance of HPCI Subsystem shall be performed as indicated in Table 4.5.1.

TABLE 4.5.1

SURVEILLANCE OF THE HPCI SUBSYSTEM

	Item	Frequency
1.	Pump Operability	Every 31 days
2.	Motor Operated Valve	Every 31 days
3.	Flow Rate Test-HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 psig when steam is being supplied to the turbine at (1000 +20, -80) psig.**	After pump maintenance and every 3 months
4.	Flow Rate Test-HPCI pump shall deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 300 psig when steam is being supplied to the turbine at 300 (+50, -100) psig.*	Prior to exceeding 350 psig following a refueling outage or an outage during which HPCI maintenance was performed.
5.	Simulated Automatic Actuation Test	Each refueling outage
6.	Logic System Functional Test	Each refueling outage

*Entry into the Startup/Hot Standby Mode is permitted provided that the required testing is successfully completed within 12 hours after reactor steam pressure is adequate to perform the test.

**Entry into the Run Mode is permitted provided that the required testing is successfully completed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

be initiated and the reactor pressure shall be reduced to 150 psig within 24 hours.

D. Automatic Pressure Relief Subsystems

1. Except as specified in 3.5.D.2 and 3.5.D.3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the five relief valves of the automatic pressure relief subsystem is made or found to be inoperable, reactor

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle the following shall be performed:

   a. A simulated automatic initiation which opens all pilot valves, and

   b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

   c. A logic system functional test shall be performed each refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

operation is permissible only during the succeeding seven (7) days provided that during such time the HPCI subsystem is operable. If the following MAPLHGR reduction factors (multipliers) are applied to Figure 3.5-1, the Automatic Pressure Relief Subsystem of ECCS shall be considered operable: (1) 0.89 for 8x8 fuel, or (2) 0.76 for 9x9 fuel.

3. From and after the date that two relief valves are found or made to be inoperable, reactor operation is permissible only during the succeeding seven days provided that during such time the HPCI subsystem is operable and the multipliers specified in 3.5.D.2 are applied.

4. If the requirements of 3.5.D.1 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to below 150 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. Whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.E.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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the requirements of 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 150 psig within 24 hours.

F. Minimum Core and Containment Cooling System Availability

1. 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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

F. Surveillance of Core and Containment Cooling System

1. 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 2, it shall not be required for Dresden Unit 3.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. When irradiated fuel is in the reactor vessel and the reactor is in cold shutdown or refuel conditions, at least two of the following pumps, with each having an operable flow path capable of taking suction from the suppression pool or the condensate storage tank and transferring the water to the reactor vessel, shall be operable except as specified in 3.5.F.4, 3.5.F.5 and 3.5.F.6 below:

   a. Two Core Spray pumps or,

   b. Two Low Pressure Coolant Injection pumps or,

   c. One Core Spray pump and one Low Pressure Coolant Injection pump.

3. With one of the pumps and/or associated flow paths required by 3.5.F.3 inoperable, restore at least two pumps and associated flow paths to operable status within 4 hours or suspend all operations with a potential for draining the reactor vessel.

4. With both of the pumps and/or associated flow paths required by 3.5.F.3 inoperable, suspend core alterations and all operations with potential for draining the reactor vessel. Restore at least one pump and associated flow path to operable status within 4 hours or establish secondary containment integrity within the next 8 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

5. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown or refuel condition, all low pressure core and containment cooling subsystems may be inoperable provided the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, the fuel pool water level is maintained above the low level alarm point, and the reactor cavity water temperature is below 140°F.

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

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

available which can be used to blank an open housing in the event of a leak.

7. When irradiated fuel is in the reactor and the vessel head is removed, work that has the potential for draining the vessel may be performed with less than 112,000 ft^3 of water in the suppression pool, provided that: 1) the total volume of water in the suppression pool, dryer separator above the shield blocks, refueling cavity, and the fuel storage pool above the bottom of the fuel pool gate is greater than 112,000 ft^3; 2) the fuel storage pool gate is removed; 3) the low pressure coolant injection and core spray systems are operable as specified in 3.5.F.3, 3.5.F.4 and 3.5.F.5; and 4) the automatic mode of the drywell sump pumps is disabled.

G. Not Used

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

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

G. Not Used

H. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray,

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10 CFR 50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified.

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(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor core cooling arise. Based on judgments of the reliability of the remaining systems; i.e., the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment a 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

When a core spray or LPCI subsystem (or pump) is inoperable, there is a requirement that other emergency core cooling systems be operable. The verification of operability, as used in this context, for these other systems means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of cooling equipment. Each set contains 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump or one LPCI pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains, a 7-day repair period was specified.

When a containment cooling subsystem (or pump) is inoperable, there is a requirement that other emergency core cooling systems be operable. The verification of operability, as used in this context, for these other systems means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI and core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

When the HPCI subsystem is inoperable, there is a requirement that other emergency core cooling systems be operable. The verification of operability, as used in this context, for these other systems means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray and LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays and LPCI. The core spray and LPCI provide sufficient flow of coolant to adequately cool the core.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Analyses have shown that only four of the five valves in the Automatic Depressurization System are required to operate. Loss of one of the relief valves does not significantly affect the pressure-relieving capability, therefore continued operation is acceptable provided the appropriate MAPLHGR reduction factor is applied to assure compliance with the 2200°F PCT limit. Loss of more than one relief valve significantly reduces the pressure relief capability of the ADS; thus; a 7-day repair period is specified with the HPCI available, and a 24-hour repair period otherwise.

When one or more relief valves is inoperable, there is a requirement that the HPCI subsystem be operable. The verification of operability, as used in this context, for HPCI means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

When the isolation condenser system is inoperable, there is a requirement that the HPCI subsystem be operable. The verification of operability, as used in this context, for HPCI means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

When the unit or shared diesel generator is inoperable, there is a requirement that other emergency core cooling systems be operable. The verification of operability, as used in this context, for these other systems means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system.

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

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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.

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.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

G. Not Used

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.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50 Appendix K considering the postulated affects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 20°F relative to the peak temperature for a typical fuel design, the limit on the average

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

planar LHGR is sufficient to assure that calculated temperatures are below the 10 CFR 50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in References (1), (2) and (3). Power operation with APLHGRs at or below those shown in Figure 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

ANF has analyzed the effects Single Loop Operation has on LOCA events (Reference 4). For breaks in the idle loop, the above Dual Loop Operation results are conservative (Reference 1). For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By applying a multiplicative 0.91 reduction factor to the results of the previous analyses, all applicable criteria are met.

J. Local Steady State LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met.

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April 1979.

(2) XN-NF-81-75 "Dresden Unit 3 LOCA Model Using the ENC EXEM Evaluation Model MAPLHGR Results"

(3) XN-NF-85-63 "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR results for 9x9 fuel", dated Septebmer 1985.

(4) ANF-84-111, "LOCA-ECCS Analysis for Dresden Units During Single Loop Operation with ANF Fuel," September 1987.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

K. Local Transient LHGR

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

L. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed as directed by the nuclear fuel vendor in accordance with station procedures.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 or 3 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2, Sheet 3 or the rated flow value, whichever is greatest.

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to all MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

M. Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig, thus, during operation even if high drywell pressure were stimulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling equipment. The verification of operability, as used in this context, for these other systems means to administratively check by examining logs or other information to determine if certain components/systems are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/system. However, if a failure, design deficiency, etc., caused the out-of-service period, then the verification of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 percent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore,

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 percent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local Steady State LHGR

The LHGR for all fuel shall be checked daily during reactor operation at greater than or equal to 25 percent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Local Transient LHGR

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. The FDLRC limit is designed to protect against centerline melting of the fuel during anticipated operational occurrences.

L. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicates that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

In addition, the reduced flow correction applied to the LCO provides margin for flow increase from low flows.

M. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

The operator will also be aware of problems in the vaults/ condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

Level	Function
a. 1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b. 3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c. 5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

shall be initiated and the reactor shall be in a cold shutdown condition in the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system subsystems shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1(a) and (b).

   a. After one of the standby gas treatment system subsystems is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment subsystem shall be operable.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (plus or minus 10%) flow through each subsystem of the standby gas treatment system for at least 10 hours with the subsystem heaters operating at rated power.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

b. If both standby gas treatment system subsystems are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

2. Performance Requirement

a. Periodic Requirements:

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

2. Performance Requirement Tests

a. At least once per 720 hours of system operation; or once per operating cycle, or every 18 months, whichever occurs first; or following painting, fire, or chemical

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

b. The system shall be shown to operate.

3. Post Maintenance Requirements

a. After any maintenance or testing that could affect the HEPA filter or HEPA filter mounting frame leak tight integrity, the results of the inplace DOP tests at 4000 cfm (plus or minus 10%) on HEPA filters shall show less than or equal to 1% DOP penetration in

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. At least once per operating cycle, or every 18 months, whichever comes first, the following conditions shall be demonstrated:

(1) Pressure drop across the combined filters of each standby gas treatment system subsystem is less than 6 inches of water at 4000 cfm (plus or minus 10%) flow rate.

(2) Operability of inlet heater at rated power.

(3) Automatic initiation of each standby gas treatment system subsystem.

3. Post Maintenance Testing

a. After any maintenance or testing that could affect the leak tight integrity of the HEPA filters, perform in-place DOP tests on the HEPA filters in accordance with Specification 3.7.B.2.a. (1).

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

accordance with Specification 3.7.B.2.a(1).

b. After any maintenance or testing that could affect the charcoal adsorber leak tight integrity, the results of in-place halogenated hydrocarbon tests at 4000 cfm (plus or minus 10%) on charcoal adsorber banks shall show less than or equal to 1% penetration in accordance with Specification 3.7.B.2.a(2).

c. The results of in-place air distribution tests shall show the air distribution is uniform within plus or minus 20% to each HEPA filter when tested initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

b. After any maintenance or testing that could affect the leak tight integrity of the charcoal adsorber banks, perform halogenated hydrocarbon tests on the charcoal adsorbers in accordance with Specification 3.7.B.2.a.(2).

c. Perform an air distribution test on the HEPA filter bank initially and after any maintenance or testing that could affect the air distribution within the standby gas treatment system. The test shall be performed at 4000 cfm (plus or minus 10%) flow rate.

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3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

C. Secondary Containment

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain a 1/4 inch of water vacuum under calm wind (less than 5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. If Specification 3.7.C.1 cannot be met, restore Secondary Containment Integrity within 4 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours and establish the conditions listed in Specification 3.7.C.1.a through d.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. At least once per operating cycle the instrument line flow check valves shall be tested for proper operation.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.00 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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.

Only one of the two standby gas treatment system subsystems is needed to cleanup the reactor building atmosphere upon containment isolation. If one subsystem is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither subsystem is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates.

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment subsystems significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

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 guage 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 subsystem is inoperable, the other subsystem will be verified to be operable. The verification of operability, as used in this context, for the other subsystem means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component/subsystem. This substantiates the availability of the operable subsystem and results in no added risk; thus, reactor operation or refueling operation can continue. If neither subsystem 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 accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The inplace test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment subsystems significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves - Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

4.7 SURVEILLANCE REQUIREMENT BASES

A. Primary Containment

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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 the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability. The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system in-place testing procedures will be established utilizing applicable sections of ANSI N510-1975 standard as a procedural guideline only. Operation of the standby gas treatment system every month for 10 hours will reduce the moisture buildup on the adsorbent. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52, Revision 1 (June 1976). The charcoal adsorber efficiency test procedures will allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of Table 3 of Regulatory Guide 1.52, Revision 1 (June 1976). The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. 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 in-place testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52, Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, air distribution to each HEPA filter, and automatic initiation of each standby gas treatment system subsystem is necessary to assure system performance capability.

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

permissible only during the succeeding seven days unless the second line is sooner placed in service providing both the Unit 3 and Unit 2/3 emergency diesel generators are operable. From and after the date that incoming power is not available from any line, reactor operation is permissible providing both the Unit 3 and Unit 2/3 emergency diesel generators are operating and all core and containment cooling systems are operable and the NRC is notified within 24 hours of the situation, the precautions to be taken during this situation, and the plans for prompt restoration of incoming power.

2. a. From and after the date that one of the diesel generators and/or its associated bus is made or found to be inoperable for any reason, except as specified in Specification 3.9.B.2.b below, reactor operation is permissible according to Specification 3.9.B.2.c and 3.9.D only during the succeeding seven days unless such diesel generator and/or bus is sooner made

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

operable, provided that during such seven days the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter and two offsite lines as specified in 3.9.A are available.

b. Specification 3.9.B.2.a shall not apply when a diesel generator has been made inoperable for a period not to exceed 1-1/2 hours for the purpose of conducting preventative maintenance. Additionally, preventative maintenance shall not be undertaken unless two offsite lines are available and the alternate diesel generator has been demonstrated to be operable.

c. 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.

3. From and after the date that one of the two 125

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

or 250V battery systems is made or found to be inoperable, except as specified in 3.9.B.4.a or b, Unit shutdown shall be initiated within 2 hours and the unit shall be in cold shutdown in 24 hours unless the failed battery can be sooner made operable.

4. a. Each 125 or 250 volt battery may be inoperable for a maximum of 7 days per operating cycle for maintenance and testing.

   b. If it is determined that a battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional 7 days per operating cycle.

C. Diesel Fuel

There shall be a minimum of 10,000 gallons of diesel fuel supply on site for each diesel.

D. Diesel Generator Operability

Whenever the reactor is in the Cold Shutdown or Refueling modes, a minimum of one diesel generator (either the Dresden 3 diesel generator or the Unit 2/3 diesel generator) shall be

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

C. Diesel Fuel

Once a month the quantity of diesel fuel available shall be logged.

Once a month a sample of diesel fuel shall be checked for quality.

D. Diesel Generator Operability

1. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue until both the diesel engine

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

operable whenever any work is being done which has the potential for draining the vessel, secondary containment is required, or a core or containment cooling system is required.

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

and the generator are at equilibrium conditions of temperature while full load output is maintained.

2. During the monthly generator test, the diesel starting air compressor shall be checked for operation and its ability to recharge air receivers.

3. During the monthly generator test, the diesel fuel oil transfer pumps shall be operated.

4. Additionally, during each refueling outage, a simulated loss of off-site power in conjunction with an ECCS initiation signal test shall be performed on the 4160 volt emergency bus by:

   (a) Verifying de-energization of the emergency buses and load shedding from the emergency buses.

3.9 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.9 SURVEILLANCE REQUIREMENT (Cont'd.)

(b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO PROPOSED LICENSE AMENDMENT TO REVISE TESTING REQUIREMENTS

OF THE EMERGENCY CORE COOLING SYSTEM (ECCS) AND STANDBY GAS

TREATMENT SYSTEMS (SGTS)

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated December 21, 1988, Commonwealth Edison Company (CECo) proposed to amend Appendix A of Provisional Operating License (POL) No. DPR-19 for Dresden Unit 2 and Facility Operating License No. DPR-25 for Dresden Unit 3 to: revise the testing requirements for other systems or subsystems of the Emergency Core Cooling System (ECCS) or Standby Gas Treatment Systems (SGTS) when one system or subsystems is inoperable; revise the operability requirements of several ECCS systems; and incorporate some administrative changes. By letter dated May 4, 1989, CECo provided supplemental information to support the proposed amendment and included two additional changes. These proposed changes which are part of the Dresden Station improvement program action plan, are consistent with similar technical specifications approved for more recently licensed BWRs and the BWR Standard Technical Specifications.

2.0 EVALUATION

A. Multiple Testing of ECCS and SBGT Systems

Present Dresden Units 2 and 3 Technical Specification Surveillance Requirements for ECCS and SBGT provide for demonstrating the operability of redundant systems or subsystems when one system or subsystem is inoperable. These requirements are as follows:

(1) One Core Spray subsystem inoperable-demonstrate operability immediately of the operable core spray subsystem and the LPCI subsystem. Demonstrate daily thereafter operability of the operable core spray subsystem.

(2) One LPCI inoperable-demonstrate operability immediately of the remaining LPCI subsystem, containment cooling subsystem, and both core spray subsystem. Demonstrate daily the operability of the operable LPCI pumps.

(3) The LPCI subsystem is inoperable-demonstrate operability immediately and daily thereafter of both core spray subsystems and the containment cooling subsystem.

(4) One containment cooling subsystem service water pump is inoperable-demonstrate operability immediately and daily thereafter the remaining components of that subsystem and the other containment cooling subsystem.

(5) One containment cooling subsystem is inoperable-demonstrate operability immediately and daily thereafter of the operable containment cooling subsystem.

(6) The HPCI subsystem is inoperable-demonstrate operability immediately of 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. Demonstrate operability daily of the motor operated isolation valves and shell side make-up system of the isolation condenser. Daily demonstration of the operability of the automatic pressure relief subsystem may be required depending on plant power level and the number of operating feedwater pumps.

(7) One of the five relief valves of the automatic pressure relief subsystem is inoperable-demonstrate the operability immediately and weekly thereafter of the HPCI subsystem.

(8) More than one relief valve of the automatic pressure relief subsystem is inoperable-demonstrate operability immediately of the HPCI subsystem.

(9) The isolation condenser system is inoperable-demonstrate operability immediately and daily thereafter of the HPCI subsystem.

(10) The unit or shared diesel generator is inoperable-demonstrate operability immediately and daily thereafter of all low pressure core cooling, the containment cooling subsystems, and the operable diesel generator.

(11) One SBGT subsystem is inoperable-demonstrate operability within 2 hours and daily thereafter of the operable SBGT subsystem.

The purpose of this proposed amendment change is to remove the redundant system testing requirements from the ECCS and SGTS sections of the Technical Specifications (Sections 4.5 and 4.7) while maintaining adequate assurance of system operability needed for accident mitigation.

The requirement for demonstrating operability of the redundant systems identified above for Dresden Units 2 and 3 was originally chosen because there was a lack of plant operating history and a lack of sufficient equipment failure data. Since that time, plant operating experience has demonstrated that testing of the redundant ECCS and SGTS when one system is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the redundant system out of service for testing creates the risk of the second system also failing and in some instances it has been observed that failures of the redundant system are related to the test itself and not an indication that the system would have failed should it have been needed. Operability of these systems can be shown by checking records to verify that valve lineups, electrical lineups and instrumentation requirements have not been changed since the last time the system was verified to be operable.

The current Standard Technical Specifications (STS) and more specifically all the technical specifications approved for recently licensed BWR's accept the philosophy of system operability based on satisfactory performance of monthly, quarterly, refueling interval, post maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing). The staff reviewed CECo's December 21, 1988 submittal and requested additional information primarily to confirm that the testing requirements for the redundant systems or subsystems contained in the existing Technical Specifications, as modified by the proposed amendments, were consistent with the requirements contained in the Standard Technical Specifications. In Attachment 2 to CECo's May 4, 1989 submittal, a comparison between the Dresden Technical Specifications and the Standard Technical Specifications was provided. The staff has reviewed this submittal and determined the proposed Technical Specifications for Dresden are consistent with the Standard Technical Specifications and those of recently licensed BWR's with regard to the testing requirements for redundant systems.

On this basis, the fact that testing of the redundant system creates the risk of the second system failing and past operational experience, the staff has determined that the revised testing requirements for the ECCS and SGTS systems and subsystems are acceptable.

In addition, other changes to Section 3.5 of the Technical Specifications have been proposed which are administrative in nature. Since these changes either clarify present requirements or promote consistency in location of requirements within the Technical Specifications (i.e. relocating all diesel generator operability requirements in one section of the Technical Specifications), the staff finds them acceptable.

During the review, a need to revise a footnote in Table 4.5.1, which waived the applicability of Specification 4.0.D and would have permitted the plant to enter into the Startup/Hot Standby Mode provided the required surveillances were successfully completed with 12 hours after reactor steam pressure is adequate to perform the test, was identified by the staff. The wording of the footnote presumed prior approval of Section 4.0.D which is also part of the Dresden Technical Specification improvement program but has not yet been submitted. CECo's May 4, 1989 submittal eliminated any reference to Section 4.0.D and included an additional footnote pertaining to entry into the Run Mode which is the same as that required for entry into the Startup/Hot Standby Mode. However, to assure that reactor operation does not continue during startup when the HPCI system testing requirements contained in Table 4.5.1 cannot be met, a proposed action statement 3.5.C.2.b has been added. The staff recognizes that some systems cannot be tested until the plant operational mode has been entered and therefore an exception to the normal Technical Specification surveillance requirements is needed for a limited time to permit the testing. These types of exceptions have been granted in the past and the staff finds them acceptable.

B. HPCI Operability Requirements

The present Technical Specification Sections 3.5.C/4.5.C require the HPCI subsystems to be operable whenever the reactor pressure is greater than 90 psig. If the HPCI is inoperable and cannot be restored within the time limits of Section 3.5.C, then the plant must be shut down and reactor pressure reduced to 90 psig. However, this present LCO requirement of 90 psig for operability of HPCI is not based on HPCI subsystem design or testing requirements. The present Surveillance Requirement in Section 4.5.C.1 requires HPCI subsystem testing to demonstrate that HPCI can deliver at least 5000 gpm against a system head corresponding to a reactor vessel pressure of 1150 to 150 psig. Since the HPCI system is designed to pump 5600 gpm into the reactor vessel within a reactor pressure range of about 1120 psig to 150 psig, the operability of the HPCI system cannot be tested at 90 psig in accordance with the current Technical Specification requirements (at pressures below 150 psig it is estimated that the flow decreases linearly to zero at 50 psig). In addition, one of the HPCI automatic isolation signals is low steam line pressure (less than 100 psig). Since the HPCI system is isolated below a steam line pressure 100 psig, the present LCO requirement of 90 psig for operability is impractical.

CECo has proposed changing the HPCI operability requirement to 150 psig to support system design flow and pressure requirements of Section 4.5.C.1 of the Technical Specifications and to provide an adequate margin to the present setpoint for system automatic isolation on low steam line pressure. The staff has reviewed this proposed change and determined it is acceptable since it corrects inconsistencies in the current Technical Specifications related to HPCI operability requirements and does not result in a decrease in safety.

CECo has also proposed to change the Surveillance Requirements in Section 4.5.C.1 to include the HPCI testing requirements (Table 4.5.1) rather than provide a reference to these requirements in the Core Spray and LPCI subsystem testing (Section 4.5.A.1). To be consistent with the standard Technical Specifications and current BWR industry practice, CECo has added a second low reactor steam pressure flow rate test to the HPCI pump flow rate testing. This second test requirement is also identified in Table 4.5.1. A test is performed every 3 months to demonstrate HPCI operability when steam is being supplied to the turbine at rated reactor pressure. The added second low pressure test is performed approximately every 18 months to demonstrate ECCS design flow when steam is being supplied to the turbine at low pressure. This proposed low pressure test will be run at a pump discharge pressure of 50 psig over reactor pressure when steam is being supplied to the turbine at 300 psig. The 350 psig upper allowable limit for testing was selected to conform with the approximate reactor pressure corresponding to the shutoff head of the low pressure coolant injection pump.

The staff has reviewed these proposed changes and determined that both the administrative changes and the additional low pressure HPCI operability test are improvements over the existing Technical Specifications and are, therefore, acceptable.

C. Automatic Pressure Relief and Isolation Condenser Operability Requirements

The present Technical Specification Sections 3.5.D (Automatic Pressure Relief) and 3.5.E (Isolation Condenser) require their respective systems to be operable whenever the reactor pressure is greater than 90 psig. CECo has proposed a Technical Specification change that would not require the Automatic Pressure Relief and the Isolation Condenser to be operable until the reactor pressure is greater than 150 psig. These changes have been proposed to preserve the consistency between the Technical Specifications for the HPCI, Automatic Depressurization System and the Isolation Condenser. Although the operability requirement is being increased from 90 to 150 psig, sufficient overlap with the low pressure systems to assure adequate core cooling will still be provided since the injection interlock for the low pressure systems is set between 300 to 350 psig. On this basis and to provide consistency between the operability requirements for these systems, the staff has concluded the proposed changes are acceptable.

D. Standby Gas Treatment System (SGTS)

The proposed changes to the SGTS Section of the Technical Specifications (Sections 3.7.B and 4.7.B) in addition to the elimination of the testing of the redundant train discussed in Section A of this Safety Evaluation are: replacing the word "circuits" with the word "subsystems;" deletion of outdated requirements for special tests in Section 4.7.B.4; and changing the test frequency for performing Surveillance Requirements 4.7.B.2a and 4.7.B.2.b.

The first two proposed changes are administrative in nature and are acceptable. The word change is editorial. The special tests are no longer required because the equipment modifications needed to allow verification of the system performance requirements are complete. The frequency of performing Surveillance requirements is presen ' stated as "once per operating cycle but not to exceed 18 months." The not to xceed 18 months requirement excludes allowances for use of the allowable standard accepted interval extensions permitted for other systems in the Technical Specifications (Definition CC). The proposed change would use the Terminology "or every 18 months whichever occurs first" which would permit the use of these interval extensions. The staff has reviewed this proposed change and, si ce it is consistent with current standard acceptable practices, finds it acceptable.

E. Secondary Containment Integrity Requirements

The proposed changes to Technical Specification Section 3.7.C on Secondary Containment integrity are: inclusion of a time frame for restoration of Secondary Containment Integrity; clarification of Definition Z on Secondary Containment Integrity; elimination of completed preoperational and first cycle operating tests and a one-time exemption which was used in 1979; and the relocation of core spray and LPCI subsystem operational requirements to Specification 3.5.A.

The first proposed change will allow 4 hours to restore Secondary Containment integrity and, if not restored, an orderly shutdown is required to at least hot shutdown within the next 12 hours and to cold shutdown within the following 24 hours. The staff has determined that these times are consistent with those of other operating nuclear plants including those that have been recently licensed and that operating experience has demonstrated these times support safe operation. The proposed orderly reactor shutdown is also consistent with the requirements of present Specification 3.0.A. The staff therefore finds this proposed change acceptable. The remaining three changes are administrative in nature and are acceptable.

F. Additional Proposed Changes in Supplemental Submittal

In CECo's May 4, 1989 submittal, two additional changes were proposed. One change, related to the Containment Cooling Service Water (CCSW) system, would add a surveillance requirement to verify that each manual, power operated or automatic valve in the flow path that is not locked, sealed or otherwise secure must be verified to be in its correct position. Since this proposed change is the same as one of the requirements to demonstrate operability of the ECCS contained in the STS and is a safety enhancement, the staff finds this acceptable. The second change, which is purely administrative adds the words "not used" next to Section 3/4.5.G and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to surveillance and operability requirements for ECCS equipment located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Part 51.22(c)(9). Pursuant to 10 CFR Part 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: Byron L. Siegel

Dated: August 10, 1989