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

June 24, 1985

Docket Nos. 50-237/249
LS05-85-06-028

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: TECHNICAL SPECIFICATIONS RELATING TO TMI ACTION ITEMS COVERED BY GENERIC LETTER 83-36

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

The Commission has issued the enclosed Amendment No. 90 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 83 to Facility Operating License No. DPR-25 for Dresden Unit 3. The amendments are in response to your application dated October 10, 1984 which was in response to Generic Letter (GL) 83-36.

The amendments add limiting conditions for operation and surveillance requirements to the Technical Specifications (TS) for certain plant modifications required by TMI Action Plan items covered by GL 83-36. The TS approved are for (1) Containment High Range Radiation Monitor (II.F.1.3); (2) Containment Pressure Monitor (II.F.1.4); (3) Containment Water Level Monitor (II.F.1.5); and (4) Containment Hydrogen Monitor (II.F.1.6). The staff agrees that, for Dresden, one item in GL 83-36, Reactor Coolant System Vents (II.B.1), does not require TS and that two others, Noble Gas Effluent Monitors (II.F.1.1) and Sampling and Analysis of Plant Effluents (II.F.1.2), have already received suitable TS. However, two more in your October 10, 1984 response, Post-Accident Sampling (II.B.3) and Control Room Habitability (III.D.3.4), require further staff review.

A Notice of Consideration of Issuance of Amendments to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on May 21, 1985 (50 FR 20972). No public comments or requests for hearing were received.

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A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

Original signed by:

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 90 to License No. DPR-19
2. Amendment No. 83 to License No. DPR-25
3. Safety Evaluation

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See next page

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Mr. Dennis L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station

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Chairman
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Morris, Illinois 60450

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Illinois Department of Nuclear Safety
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 90
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated October 10, 1984 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.

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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 Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 90, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 24, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 90

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

Revise 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
ii	ii
vii	vii
viii	viii
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3/4.2-17	3/4.2-17
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B 3/4.2-25	3/4.2-25*
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B 3/4.2-29	B 3/4.2-29*
B 3/4.2-30	B 3/4.2-30*
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B 3/4.2-33	B 3/4.2-33
B 3/4.2-34	B 3/4.2-34*
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3.2 LIMITING CONDITION FOR OPERATION (CONT'D)

in the fuel storage pool and during refueling or fuel movement operations.

2. One of the two refueling floor radiation monitors may be inoperable for 24 hours. If the inoperable monitor is not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The trip setting for the refueling floor radiation monitors shall be set at a less than or equal to 100mr/hr.

4. Upon loss of both refueling floor radiation monitors while in use, the reactor building ventilation system shall be isolated and the standby gas treatment operated.

E. Post Accident Instrumentation

The limiting conditions for operation for the instrumentation, which is read out in the control room, required for post accident monitoring are given in Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS (CONT'D)

treatment system initiation shall be performed at least each operating cycle.

E. Post Accident Instrumentation

Post accident instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.4.

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Table 3.2.6
Post Accident Monitoring Instrumentation Requirements

Minimum Number of Operable Channels (1)	Parameter	Instrument Readout Location Unit 2	Number Provided	Instrument Range
1	Reactor Pressure	902-5	1 2	0-1500 psig 0-1200 psig
1	Reactor Water Level	902-3	2	-340 to +60 inches
1	Torus Water Temperature	902-4	2	0-200°F
2 (3)	Torus Water Level Indicator	902-3 902-3	1 1	-25 to +25 inches -7 to +3 inches (narrow range)
		902-2	2	0-30 ft (wide range)
	Torus Water Local Sight Glass		1	18 inch range (narrow range)
1 (4)	Torus Pressure	902-5	1	-2.45-5 psig
2	Drywell Pressure	902-5 902-3 902-3	1 1 2	0-5 psig 0-75 psig 0-250 psig
2	Drywell Temperature	902-21	6	0-600°F
2	Neutron Monitoring	902-5	4	0.1-10^6 CPS
1 (4)	Torus to Drywell Differential Pressure	902-3	2	0-3 psid
1	Drywell Radiation Monitor	902-55,56	2	1 to 10^8 R/hr
2/valve (2)	Main Steam RV Position, Acoustic Monitor	902-21	1 per valve	N/A
	Main Steam RV Position, Temperature Monitor	902-21	1 per valve	0-600°F
2/valve (2)	Main Steam SV Position, Acoustic Monitor	902-21	1 per valve	N/A
	Main Steam SV Position, Temperature Monitor	902-21	1 per valve	0-600°F
1 (5)	Drywell Hydrogen Concentration	902-55 902-56	2	0-10%

Notes: (See Next Page)

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Table 3.2.6

Notes

1. From and after the date that a parameter is reduced to the minimum number of channels, continued operation is not permissible beyond thirty (30) days unless such instrumentation is sooner made operable. In the event that all indications of a parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours. See notes 2, 3, 4 and 5 for exceptions to this requirement.

2. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a cold shutdown condition for longer than 12 hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty (30) days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in twenty-four (24) hours.

3. From and after the date that this parameter is reduced to either one narrow-range indication or one wide-range indication, continued reactor operation is not permissible beyond thirty (30) days unless such instrument is sooner made operable. In the event that either all narrow-range indication or all wide-range indication is disabled, continued reactor operation is not permissible beyond seven (7) days unless such instruments are sooner made operable. In the event that all indication for this parameter is disabled, and such indication cannot be restored in six (6) hours, an orderly shutdown shall be intitiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

4. From and after the date that one of these parameters becomes inoperable, continued operation is not permissible beyond thirty (30) days unless such instrumentation is sooner made operable. In the event that all indication of these parameters is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be intitiated and the reactor shall be in cold shutdown in twenty-four (24) hours.

5. From and after the date that one of the drywell hydrogen monitors becomes inoperable, continued reactor operation is permissible.

   a. If both drywell hydrogen monitors are inoperable, continued reactor operation is permissible for up to 30 days provided that during this time the HRSS hydrogen monitoring capability for the drywell is operable.

   b. If all drywell hydrogen monitoring capability is lost, continued reactor operations is permissible for up to 7 days.

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Table 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
ECCS Instrumentation			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1) (13)	(13)	None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refuel Outage	Once/3 months
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
8. Degraded Voltage Emergency Bus	Refueling Outage (10)	Refuel Outage	Monthly
Rod Blocks			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refuel Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM Detector Not Fully Inserted in the Core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refuel Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instrument Volume Level High	Once/3 Months (9)	None	None
Containment Monitoring			
1. Pressure			
a. Minus 5 in. Hg to plus 5 psig Indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig Indicator	None	Once/3 Months	None
2. Temperature	None	Refuel Outage	Once/Day
3. Drywell-Torus Differential Pressure (5) (6) (0-3 psid)	None	Once/6 Months (Two Channels Operable) Once/Month (One Channel Operable)	None
4. Torus Water Level (5) (6)	None	Once/6 Months	
a. Plus or minus 25 in. Wide Range Indicator			
b. 18 in. Sight Glass			
Safety/Relief Valve Monitoring			
1. Safety/Relief Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
2. Safety/Relief Valve Position Indicator (Temperature Monitor (8)	None	Once every 18 months	Once Per 31 Days
3. Safety Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
4. Safety Valve Position Indicator (Temperature Monitor) (8)	None	Once every 18 months	Once Per 31 Days

(Table cont'd next page)

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Table 4.2.1
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS (Cont'd)

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refuel Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
Isolation Condenser Isolation			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI Isolation			
1. Steam Line High Flow	(1) (11) (12)	(11) (12)	None
2. Steam Line Area High Temperature	Refueling Outage	Refuel Outage	None
3. Low Reactor Pressure	(1) (13)	(13)	None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day

NOTES: (For Table 4.2.1)

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 2.

2. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

(Cont'd. next page)

NOTES: (For Table 4.2.1) (Cont'd.)

6. From and after the date that one of these parameters (...either drywell-torus differential pressure or torus water level indication) is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters (...either drywell-torus differential pressure or torus water level) is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

7. Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular Safety/Relief Valve.

8. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a cold shutdown condition for more than seventy-two hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in 24 hours.

9. The functional test of the Scram Discharge Volume thermal switches is not applicable; i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

10. Functional test shall include verification of the second level undervoltage (degraded voltage) timer bypass and shall verify operation of the degraded voltage 5-minute timer and inherent 7-second timer.

11. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

12. Trip units are functionally tested monthly (staggered one channel out of four every week). A calibration of the trip units is to be performed concurrent with the functional testing.

13. Trip units are functionally tested monthly (staggered one division out of two every two weeks). A calibration of the trip units is to be performed concurrent with the functional testing.

TABLE 4.2.2

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)(7)	Calibration (1)(7)(3)(4)	Functional Test (1)(2)(7)	Source Check (1)
Liquid Radwaste Effluent Gross Activity Monitor	D	R	Q (6)	(5)
Service Water Effluent Gross Activity Monitor	D	R	Q (6)	R
Tank Level Indicating Device				
a. A Waste Sample Tank	D	R	Q	N/A
b. B Waste Sample Tank	D	R	Q	N/A
c. C Waste Sample Tank	D	R	Q	N/A
d. A Floor Drain Sample Tank	D	R	Q	N/A
e. B Floor Drain Sample Tank	D	R	Q	N/A
f. Waste Surge Tank	D	R	Q	N/A

Notes:
(See Next Page)

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TABLE 4.2.2 (Notes)

1. D = Once per 24 hours
   M = Once per 31 days
   Q = Once per 92 days
   R = Once per refueling outage
   S = Once per 6 months

2. The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

   a. Instrument indicates levels above the alarm setpoint.
   b. Circuit Failure.
   c. Instrument indicates a downscale failure.
   d. Instrument controls not set in OPERATE mode.

3. Calibration shall include performance of a functional test.

4. Calibration shall include performance of a source check.

5. Source check shall consist of observing instrument response during a discharge.

6. Functional test may be performed by using trip check and test circuitry associated with the monitor chassis.

7. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Calibration shall be performed once per refueling outage and not more than once every 18 months. Instrument checks shall be performed at least once a day during those periods when the instruments are required to be operable.

TABLE 4.2.3

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)(6)	Calibration (1)(6)(3)	Function Test (1)(4)(2)(6)	Source Check (1)
SJAE Radiation Activity Monitor	D	R	Q	R
Reactor Bldg Vent Particulate and Iodine Sampler	D (4)	N/A	N/A	N/A
Reactor Bldg Vent Exhaust Duct Radiation Monitor	D	R	Q	Q
Reactor Bldg Vent SPING Noble Gas Monitor Lo, Mid, High Range	D	R	Q	M
Main Chimney Noble Gas Activity Monitor	D	R	Q	M
Main Chimney SPING Noble Gas Monitor Lo, Mid, High Range	D	R	Q	M
Main Chimney Particulate and Iodine Sampler	D (4)	N/A	N/A	N/A
Main Chimney Flow Rate Monitor	D	R	Q	N/A
Main Chimney Sampler Flow Rate Monitor	D	R	Q (5)	N/A
Reactor Bldg Vent Flow Rate Monitor	D	R	Q	N/A
Reactor Bldg Sampler Flow Rate Monitor	D	R	Q (5)	N/A

Notes: (See Next Page)

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TABLE 4.2.3 (Notes)

1. D = Once per 24 hours
   M = Once per 31 days
   Q = Once per 92 days
   R = Once per refueling outage

2. The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

   a. Instrument indicates levels above the alarm setpoint.
   b. Circuit Failure.
   c. Instrument indicates a downscale failure.
   d. Instrument controls not set in OPERATE mode.

3. Calibration shall include performance of a functional test.

4. Instrument check to verify operability of sampler; that the sampler is in place and functioning properly.

5. Function Test shall be performed on local switches providing low flow alarm.

6. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Calibration shall be performed once per refueling outage and not more than once every 18 months. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

Table 4.2.4

Post Accident Monitoring Instrumentation Surveillance Requirements

Minimum Number of Operable Channels	Parameter	Instrument Readout Location Unit 2	Calibration	Instrument Check
1	Reactor Pressure	902-5	Once Every 6 Months	Once Per Day
1	Reactor Water Level	902-3	Once Every 6 Months	Once Per Day
1	Torus Water Temperature	902-4	Once Every 12 Months	Once Per Day
2	Torus Water Level Indicator (Narrow Range)	902-3	Once Every 6 Months	Once Per Day
	(Sight Glass)		N/A	None
	(Wide Range)	902-2	Once Every 12 Months	Once Per 31 Days
1	Torus Pressure	902-3,5	Once Every 3 Months	Once Per Day
1	Torus to Drywell Differential Pressure	902-3	Once Every 6 Months	Once Per Day
2	Drywell Pressure			
	(0-5 psig)	902-5	Once Every 3 Months	Once Per Day
	(0-75 psig)	902-3	Once Every 3 Months	Once Per 31 Days
	(0-250 psig)	902-3	Once Every Refuel	Once Per 31 Days
2	Drywell Temperature	902-21	Once Every Refuel	Once Per Day
2	Neutron Monitoring	902-5	Once Every 3 Months	Once Per Day
1	Drywell Radiation Monitor	902-55,56	Once Every Refuel (2)	Once Per 31 Days
2/Valve	Main Steam RV Position, Temperature Monitor	902-21	Once Every Refuel	Once Per 31 Days
	Main Steam RV Position, Acoustic Monitor		(1)	
2/Valve	Main Steam SV Position, Temperature Monitor	902-21	Once Every Refuel	Once Per 31 Days
	Main Steam SV Position, Acoustic Monitor		(1)	Once Per 31 Days
1	Drywell Hydrogen Concentration	902-55 902-56	Once Every 3 Months	Once Per 31 Days

Notes: (See Next Page)

Table 4.2.4 (Notes)

Notes

1. Calibration of Acoustic Monitors shall consist of verifying the instrument threshold levels, and will be performed monthly.

   Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular safety or relief valve.

2. Calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr; and a one-point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

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3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are 1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and 2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation which serves the condition for which isolation is required (this instrumentation is shown in Table 3.2.1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor level instrumentation is set to trip at greater than 8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the

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

top of active fuel. Retrofit 8 X 8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, - 59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group I primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems, starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary isolation are initiated and in time to meet the above criteria. The instrumentation also covers the full spectrum of breaks and meets the above criteria.

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here, also Group 2 isolation valves include the drywell vent, purge and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

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

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

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

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times full power background for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steamline trip setting is less than or equal to 3 times full power background with hydrogen addition, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. (Ref. Section 14.2.1.7 SAR) The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the NRC.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided to provide protection against a pressure regulator malfunction which would cause the control

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

and/or bypass valves to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500 degrees F; thus, there are no fission products available for release other than those in the reactor water. (Ref. Section 11.2.3 SAR)

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32 inches of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR fuel cladding integrity safety limit. The trip logic for this function is 1 out of n, e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. This time period is only approximately 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at

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

reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the MCPR fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the MCPR fuel cladding integrity safety limit. Thus, the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

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

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (Ref. SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of less than or equal to 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the post accident condition is listed in Table 3.2.6. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident. Based on this information he can make logical decisions regarding post accident recovery.

The specifications allow for post accident instrumentation to be out of service for a period of 30 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 30-day period, and on engineering judgement.

The radioactive liquid and gaseous effluent instrumentation is provided to monitor the release of radioactive materials in liquid and gaseous effluents during releases. The alarm setpoints for the instruments are provided to ensure that the alarms will occur prior to exceeding the limits of 10 CFR 20.

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4.2 SURVEILLANCE REQUIREMENT BASES

The instrumentation listed in Table 4.2.1 will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goal of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (see note 7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = (2t/r)^{1/2}$$

Where:

i = optimum interval between tests

t = the time the trip contacts are disabled from performing their function while the test is in progress

r = The expected failure rate of the relays

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = [2(0.5)/10^{-6}]^{1/2} = 1 \times 10^{3} \text{ hours}$$

$$= \text{approximately 40 days}$$

<u>For additional margin a test</u> interval of once per month will be used initially.

Note:

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

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4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time. Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one out of n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by curve No. 1 of Figure 4.2.2 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and that 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1 out of 2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested and restored, and then immediately following, the second channel be bypassed, tested and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. More than one channel should not be bypassed for testing at any one time.

The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog system, including reactor low water level, has been established in Licensing Topical Report NEDO-21617-A (December, 1978).

For instruments 2(3)-2389A, B, C, D, the one-of-two-taken-twice logic exists, and NEDO-21617-A states that each trip unit be subjected to a calibration/test frequency (staggered one channel out of four per week) of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

For instruments 2(3)-263-73A, 73B and 2(3)-2352, 2353, the logic downstream of the output relay contacts exhibits a one-out-of-two logic and, by utilizing the Availability Criteria identified in NEDO-21617-A, each of these trip units should also be subjected to a calibration/test frequency (staggered one division out of two per two weeks) of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks applies here also and were used to arrive at the functional testing frequency.

Based on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every three months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the post accident condition will be tested and calibrated at regularly scheduled intervals. The basis for the calibration and testing of this instrumentation is the same as was discussed above for Protective Instrumentation in Table 4.2.4.

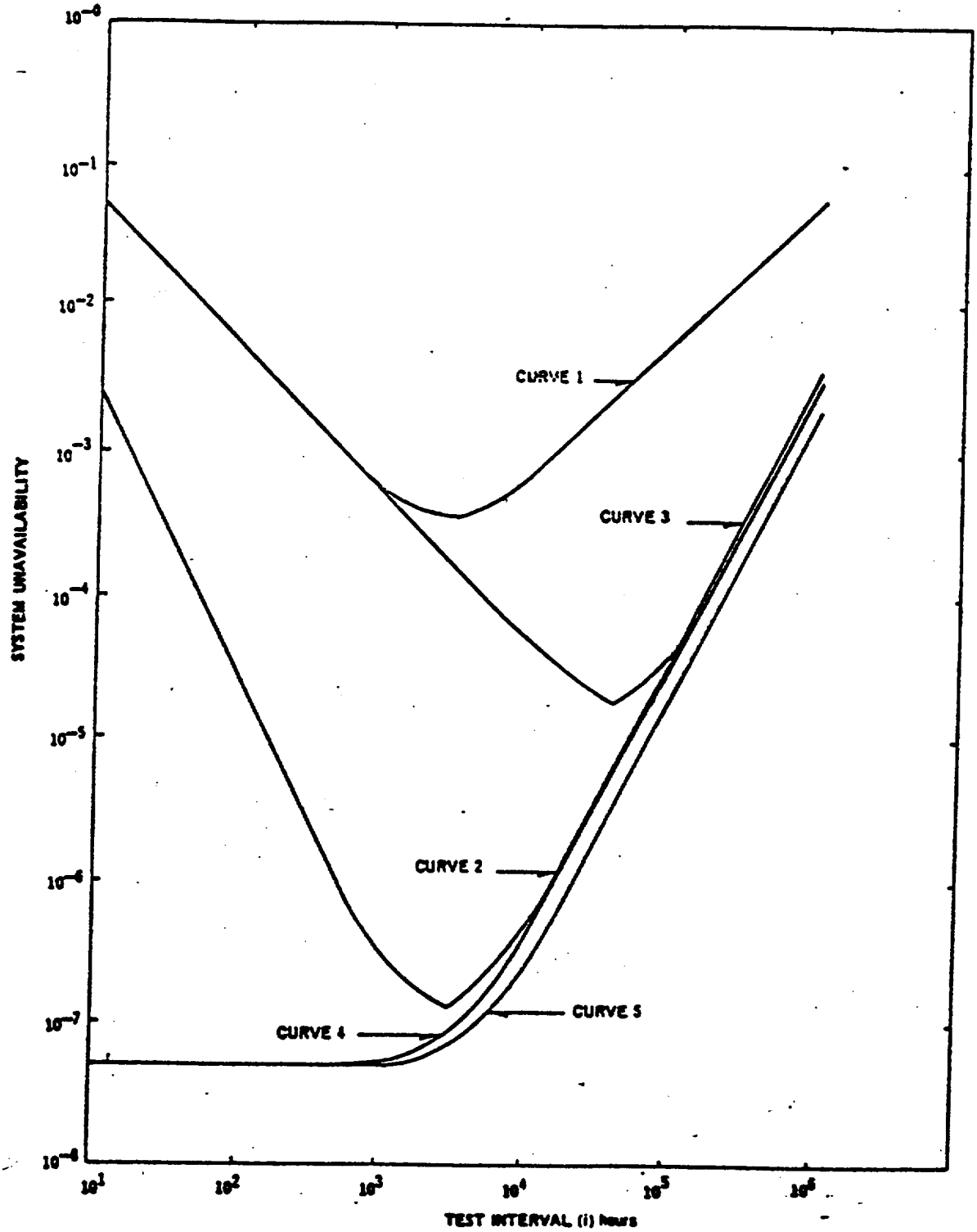

Figure 4.2.2
TEST INTERVAL VS. SYSTEM UNAVAILABILITY

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
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 October 10, 1984 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. 83, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 24, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

Revise 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
ii	ii
vii	vii
viii	viii
3/4.2-3	3/4.2-3
3/4.2-17	3/4.2-17
3/4.2-18	3/4.2-18
3/4.2-19	3/4.2-19*
3/4.2-20	3/4.2-20*
3/4.2-21	3/4.2-21*
3/4.2-22	3/4.2-22*
3/4.2-23	3/4.2-23*
B 3/4.2-24	3/4.2-24*
B 3/4.2-25	3/4.2-25*
B 3/4.2-26	3/4.2-26
B 3/4.2-27	3/4.2-27
B 3/4.2-28	B 3/4.2-28*
B 3/4.2-29	B 3/4.2-29*
B 3/4.2-30	B 3/4.2-30*
B 3/4.2-31	B 3/4.2-31*
B 3/4.2-32	B 3/4.2-32*
B 3/4.2-33	B 3/4.2-33
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	B 3/4.2-35*
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* Pagination change only

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3.2 LIMITING CONDITION FOR OPERATION (CONT'D)

in the fuel storage pool and during refueling or fuel movement operations.

2. One of the two refueling floor radiation monitors may be inoperable for 24 hours. If the inoperable monitor is not restored to service in this time, the reactor building ventilation system shall be isolated and the standby gas treatment operated until repairs are complete.

3. The trip setting for the refueling floor radiation monitors shall be set at a less than or equal to 100mr/hr.

4. Upon loss of both refueling floor radiation monitors while in use, the reactor building ventilation system shall be isolated and the standby gas treatment operated.

E. Post Accident Instrumentation

The limiting conditions for operation for the instrumentation, which is read out in the control room, required for post accident monitoring are given in Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS (CONT'D)

treatment system initiation shall be performed at least each operating cycle.

E. Post Accident Instrumentation

Post accident instrumentation shall be functionally tested and calibrated as indicated in Table 4.2.4.

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Table 3.2.6
Post Accident Monitoring Instrumentation Requirements

Minimum Number of Operable Channels (1)	Parameter	Instrument Readout Location Unit 3	Number Provided	Instrument Range
1	Reactor Pressure	903-5	1 2	0-1500 psig 0-1200 psig
1	Reactor Water Level	903-3	2	-340 to +60 inches
1	Torus Water Temperature	903-4	2	0-200°F
2 (3)	Torus Water Level Indicator	903-3 903-3	1 1	-25 to +25 inches -7 to +3 inches (narrow range)
		903-2	2	0-30 ft (wide range)
	Torus Water Local Sight Glass		1	18 inch range (narrow range)
1 (4)	Torus Pressure	903-5	1	-2.45-5 psig
2	Drywell Pressure	903-5 903-3 903-3	1 1 2	0-5 psig 0-75 psig 0-250 psig
2	Drywell Temperature	903-21	6	0-600°F
2	Neutron Monitoring	903-5	4	$0.1-10^6$ CPS
1 (4)	Torus to Drywell Differential Pressure	903-3	2	0-3 psid
1	Drywell Radiation Monitor	903-55,56	2	1 to 10^8 R/hr
2/valve (2)	Main Steam RV Position, Acoustic Monitor	903-21	1 per valve	N/A
	Main Steam RV Position, Temperature Monitor	903-21	1 per valve	0-600°F
2/valve (2)	Main Steam SV Position, Acoustic Monitor	903-21	1 per valve	N/A
	Main Steam SV Position, Temperature Monitor	903-21	1 per valve	0-600°F
1 (5)	Drywell Hydrogen Concentration	903-55 903-56	2	0-10%

Notes: (See Next Page)

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Table 3.2.6

Notes

1. From and after the date that a parameter is reduced to the minimum number of channels, continued operation is not permissible beyond thirty (30) days unless such instrumentation is sooner made operable. In the event that all indications of a parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours. See notes 2, 3, 4 and 5 for exceptions to this requirement.

2. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a cold shutdown condition for longer than 12 hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty (30) days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in twenty-four (24) hours.

3. From and after the date that this parameter is reduced to either one narrow-range indication or one wide-range indication, continued reactor operation is not permissible beyond thirty (30) days unless such instrument is sooner made operable. In the event that either all narrow-range indication or all wide-range indication is disabled, continued reactor operation is not permissible beyond seven (7) days unless such instruments are sooner made operable. In the event that all indication for this parameter is disabled, and such indication cannot be restored in six (6) hours, an orderly shutdown shall be intitiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

4. From and after the date that one of these parameters becomes inoperable, continued operation is not permissible beyond thirty (30) days unless such instrumentation is sooner made operable. In the event that all indication of these parameters is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be intitiated and the reactor shall be in cold shutdown in twenty-four (24) hours.

5. From and after the date that one of the drywell hydrogen monitors becomes inoperable, continued reactor operation is permissible.

   a. If both drywell hydrogen monitors are inoperable, continued reactor operation is permissible for up to 30 days provided that during this time the HRSS hydrogen monitoring capability for the drywell is operable.

   b. If all drywell hydrogen monitoring capability is lost, continued reactor operation is permissible for up to 7 days.

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Table 4.2.1
MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
ECCS Instrumentation			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	Once/Day
2. Drywell High Pressure	(1)	Once/3 Months	None
3. Reactor Low Pressure	(1)	Once/3 Months	None
4. Containment Spray Interlock			
a. 2/3 Core Height	(1)		None
b. Containment High Pressure	(1)	Once/3 Months	None
5. Low Pressure Core Cooling Pump Discharge	(1)	Once/3 Months	None
6. Undervoltage Emergency Bus	Refueling Outage	Refuel Outage	Once/3 months
7. Sustained High Reactor Pressure	(1)	Once/3 Months	None
8. Degraded Voltage Emergency Bus	Refueling Outage (10)	Refuel Outage	Monthly
Rod Blocks			
1. APRM Downscale	(1) (3)	Once/3 Months	None
2. APRM Flow Variable	(1) (3)	Refuel Outage	None
3. APRM Upscale (Startup/Hot Standby)	(2) (3)	(2) (3)	(2)
4. IRM Upscale	(2) (3)	(2) (3)	(2)
5. IRM Downscale	(2) (3)	(2) (3)	(2)
6. IRM Detector Not Fully Inserted in the Core	(2)	N/A	None
7. RBM Upscale	(1) (3)	Refuel Outage	None
8. RBM Downscale	(1) (3)	Once/3 Months	None
9. SRM Upscale	(2) (3)	(2) (3)	(2)
10. SRM Detector Not in Startup Position	(2) (3)	(2) (3)	(2)
11. Scram Instrument Volume Level High	Once/3 Months (9)	None	None
Containment Monitoring			
1. Pressure			
a. Minus 5 in. Hg to plus 5 psig Indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig Indicator	None	Once/3 Months	None
2. Temperature	None	Refuel Outage	Once/Day
3. Drywell-Torus Differential Pressure (5) (6) (0-3 psid)	None	Once/6 Months (Two Channels Operable) Once/Month (One Channel Operable)	None
4. Torus Water Level (5) (6)	None	Once/6 Months	
a. Plus or minus 25 in. Wide Range Indicator			
b. 18 in. Sight Glass			
Safety/Relief Valve Monitoring			
1. Safety/Relief Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
2. Safety/Relief Valve Position Indicator (Temperature Monitor (8)	None	Once every 18 months	Once Per 31 Days
3. Safety Valve Position Indicator (Acoustic Monitor) (8)	(7)	None	Once Per 31 Days
4. Safety Valve Position Indicator (Temperature Monitor) (8)	None	Once every 18 months	Once Per 31 Days

(Table cont'd next page)

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Instrument Channel	Instrument Functional Test	Calibration	Instrument Check
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refuel Outage	None
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1)	Once/3 Months	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months (4)	Once/Day
Isolation Condenser Isolation			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI Isolation			
1. Steam Line High Flow	(1) (11)	(11)	None
2. Steam Line Area High Temperature	Refueling Outage	Refuel Outage	None
3. Low Reactor Pressure	(1)		None
Reactor Building Vent Isolation and SBGTS Initiation			
1. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day

NOTES: (For Table 4.2.1)

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 3.

2. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

(Cont'd next page)

NOTES: (For Table 4.2.1) (Cont'd.)

5. A minimum of two channels is required.

6. From and after the date that one of these parameters (...either drywell-torus differential pressure or torus water level indication) is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters (...either drywell-torus differential pressure or torus water level) is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

7. Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular Safety/Relief Valve.

8. If the number of position indicators is reduced to one indication on one or more valves, continued operation is permissible; however, if the reactor is in a shutdown conditionfor more than seventy-two hours, it may not be started up until all position indication is restored. In the event that all position indication is lost on one or more valves and such indication cannot be restored in thirty days, an orderly shutdown shall be initiated, and the reactor shall be depressurized to less than 90 psig in 24 hours.

9. The Functional Test of the Scram Discharge Volume float switch shall include actuation of the switch using a water column.

10. Functional test shall include verification of the second level undervoltage (degraded voltage) timer bypass and shall verify operation of the degraded voltage 5-minute timer and inherent 7-second timer.

11. Verification of time delay setting between 3 and 9 seconds shall be performed during each refueling outage.

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TABLE 4.2.2

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)(7)	Calibration (1)(7)(3)(4)	Functional Test (1)(2)(7)	Source Check (1)
Liquid Radwaste Effluent Gross Activity Monitor	D	R	Q (6)	(5)
Service Water Effluent Gross Activity Monitor	D	R	Q (6)	R
Tank Level Indicating Device				
a. A Waste Sample Tank	D	R	Q	N/A
b. B Waste Sample Tank	D	R	Q	N/A
c. C Waste Sample Tank	D	R	Q	N/A
d. A Floor Drain Sample Tank	D	R	Q	N/A
e. B Floor Drain Sample Tank	D	R	Q	N/A
f. Waste Surge Tank	D	R	Q	N/A

Notes:
(See Next Page)

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TABLE 4.2.2 (Notes)

1. D = Once per 24 hours
   M = Once per 31 days
   Q = Once per 92 days
   R = Once per refueling outage
   S = Once per 6 months

2. The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

   a. Instrument indicates levels above the alarm setpoint.
   b. Circuit Failure.
   c. Instrument indicates a downscale failure.
   d. Instrument controls not set in OPERATE mode.

3. Calibration shall include performance of a functional test.

4. Calibration shall include performance of a source check.

5. Source check shall consist of observing instrument response during a discharge.

6. Functional test may be performed by using trip check and test circuitry associated with the monitor chassis.

7. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Calibration shall be performed once per refueling outage and not more than once every 18 months. Instrument checks shall be performed at least once a day during those periods when the instruments are required to be operable.

TABLE 4.2.3

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check (1)(6)	Calibration (1)(6)(3)	Function Test (1)(4)(2)(6)	Source Check (1)
SJAE Radiation Activity Monitor	D	R	Q	R
Reactor Bldg Vent Particulate and Iodine Sampler	D (4)	N/A	N/A	N/A
Reactor Bldg Vent Exhaust Duct Radiation Monitor	D	R	Q	Q
Reactor Bldg Vent SPING Noble Gas Monitor Lo, Mid, High Range	D	R	Q	M
Main Chimney Noble Gas Activity Monitor	D	R	Q	M
Main Chimney SPING Noble Gas Monitor Lo, Mid, High Range	D	R	Q	M
Main Chimney Particulate and Iodine Sampler	D (4)	N/A	N/A	N/A
Main Chimney Flow Rate Monitor	D	R	Q	N/A
Main Chimney Sampler Flow Rate Monitor	D	R	Q (5)	N/A
Reactor Bldg Vent Flow Rate Monitor	D	R	Q	N/A
Reactor Bldg Sampler Flow Rate Monitor	D	R	Q (5)	N/A

Notes:
(See Next Page)

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TABLE 4.2.3 (Notes)

1. D = Once per 24 hours
   M = Once per 31 days
   Q = Once per 92 days
   R = Once per refueling outage

2. The Instrument Functional Test shall also demonstrate that control room alarm annunciation occurs, if any of the following conditions exist, where applicable.

   a. Instrument indicates levels above the alarm setpoint.
   b. Circuit Failure.
   c. Instrument indicates a downscale failure.
   d. Instrument controls not set in OPERATE mode.

3. Calibration shall include performance of a functional test.

4. Instrument check to verify operability of sampler; that the sampler is in place and functioning properly.

5. Function Test shall be performed on local switches providing low flow alarm.

6. Function test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Calibration shall be performed once per refueling outage and not more than once every 18 months. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

Table 4.2.4

Post Accident Monitoring Instrumentation Surveillance Requirements

Minimum Number of Operable Channels	Parameter	Instrument Readout Location Unit 3	Calibration	Instrument Check
1	Reactor Pressure	903-5	Once Every 6 Months	Once Per Day
1	Reactor Water Level	903-3	Once Every 6 Months	Once Per Day
1	Torus Water Temperature	903-4	Once Every 12 Months	Once Per Day
2	Torus Water Level Indicator (Narrow Range)	903-3	Once Every 6 Months	Once Per Day
	(Sight Glass)		N/A	None
	(Wide Range)	903-2	Once Every 12 Months	Once Per 31 Days
1	Torus Pressure	903-3,5	Once Every 3 Months	Once Per Day
1	Torus to Drywell Differential Pressure	903-3	Once Every 6 Months	Once Per Day
2	Drywell Pressure			
	(0-5 psig)	903-5	Once Every 3 Months	Once Per Day
	(0-75 psig)	903-3	Once Every 3 Months	Once Per 31 Days
	(0-250 psig)	903-3	Once Every Refuel	Once Per 31 Days
2	Drywell Temperature	903-21	Once Every Refuel	Once Per Day
2	Neutron Monitoring	903-5	Once Every 3 Months	Once Per Day
1	Drywell Radiation Monitor	903-55,56	Once Every Refuel (2)	Once Per 31 Days
2/Valve	Main Steam RV Position, Temperature Monitor	903-21	Once Every Refuel	Once Per 31 Days
	Main Steam RV Position, Acoustic Monitor		(1)	
2/Valve	Main Steam SV Position, Temperature Monitor	903-21	Once Every Refuel	Once Per 31 Days
	Main Steam SV Position, Acoustic Monitor		(1)	Once Per 31 Days
1	Drywell Hydrogen Concentration	903-55 903-56	Once Every 3 Months	Once Per 31 Days

Notes: (See Next Page)

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Table 4.2.4 (Notes)

Notes

1. Calibration of Acoustic Monitors shall consist of verifying the instrument threshold levels, and will be performed monthly.

   Functional tests will be conducted before startup at the end of each refueling outage or after maintenance is performed on a particular safety or relief valve.

2. Calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr; and a one-point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

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3.2 LIMITING CONDITION FOR OPERATION BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are 1) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and 2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation which serves the condition for which isolation is required (this instrumentation is shown in Table 3.2.1). Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor level instrumentation is set to trip at greater than 8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the

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

top of active fuel. Retrofit 8 X 8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, - 59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group I primary containment isolation valves (reference SAR Section 7.7.2.2) and also activates the ECC subsystems, starts the emergency diesel generator, and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be exceeded. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary isolation are initiated and in time to meet the above criteria. The instrumentation also covers the full spectrum of breaks and meets the above criteria.

The high-drywell pressure instrumentation is a backup to the water level instrumentation and, in addition to initiating ECCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here, also Group 2 isolation valves include the drywell vent, purge and sump isolation valves. High-drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions, and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of Group 1 primary system isolation valves.

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

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

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

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times full power background for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steamline trip setting is less than or equal to 3 times full power background with hydrogen addition, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. (Ref. Section 14.2.1.7 SAR) The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the NRC.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided to provide protection against a pressure regulator malfunction which would cause the control

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

and/or bypass valves to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500 degrees F; thus, there are no fission products available for release other than those in the reactor water. (Ref. Section 11.2.3 SAR)

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32 inches of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR fuel cladding integrity safety limit. The trip logic for this function is 1 out of n, e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. This time period is only approximately 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at

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

reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the MCPR fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the MCPR fuel cladding integrity safety limit. Thus, the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

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

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met (Ref. SAR Section 6.2.6.3). The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided on the refueling floor which initiate isolation of the reactor building and operation of the standby gas treatment systems. The trip logic is one out of two. Trip settings of less than or equal to 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the post accident condition is listed in Table 3.2.6. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident. Based on this information he can make logical decisions regarding post accident recovery.

The specifications allow for post accident instrumentation to be out of service for a period of 30 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 30-day period, and on engineering judgement.

The radioactive liquid and gaseous effluent instrumentation is provided to monitor the release of radioactive materials in liquid and gaseous effluents during releases. The alarm setpoints for the instruments are provided to ensure that the alarms will occur prior to exceeding the limits of 10 CFR 20.

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4.2 SURVEILLANCE REQUIREMENT BASES

The instrumentation listed in Table 4.2.1 will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goal of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (see note 7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = (2t/r)^{1/2}$$

Where:

i = optimum interval between tests

t = the time the trip contacts are disabled from performing their function while the test is in progress

r = The expected failure rate of the relays

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = [2(0.5)/10^{-6}]^{1/2} = 1 \times 10^{3} \text{ hours}$$

$$= \text{approximately 40 days}$$

For additional margin a test interval of once per month will be used initially.

Note:
(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

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4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time. Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one out of n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by curve No. 1 of Figure 4.2.2 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and that 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1 out of 2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

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4.2 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

A more usual case is that the testing is done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed tested and restored. Then immediately following, the second channel will be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and
2. More than one channel should not be bypassed for testing at any one time.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks applies here also and were used to arrive at the functional testing frequency.

Based on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every three months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the post accident condition will be tested and calibrated at regularly scheduled intervals. The basis for the calibration and testing of this instrumentation is the same as was discussed above for Protective Instrumentation in Table 4.2.4.

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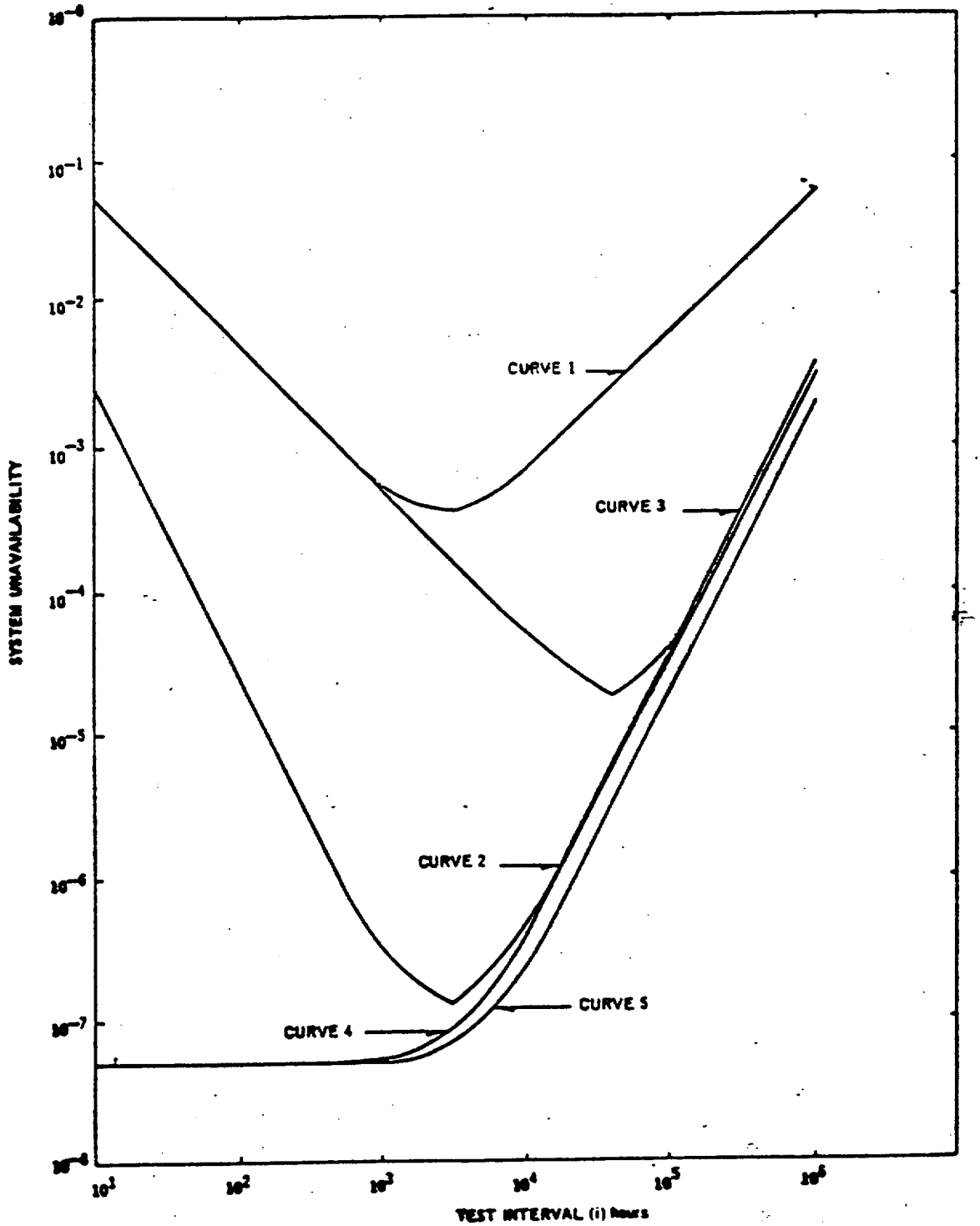

Figure 4.2.2

TEST INTERVAL VS. SYSTEM UNAVAILABILITY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237/249

1.0 INTRODUCTION

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications are required. A number of items which require Technical Specifications (TS) were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of Technical Specifications for all of these items in Generic Letter 83-36. Generic Letter 83-36 was issued to all Boiling Water Reactor (BWR) licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

1. review their facility's Technical Specifications to determine if they were consistent with the guidance provided in the Generic Letter, and

2. submit an application for a license amendment where deviations or absence of Technical Specifications were found.

By letter dated October 10, 1984, Commonwealth Edison Company (the licensee) responded to Generic Letter 83-36 by submitting a Technical Specification change request for Dresden Units 2 and 3. This evaluation covers the following TMI Action Plan items:

1. Containment High-Range Radiation Monitor (II.F.1.3)
2. Containment Pressure Monitor (II.F.1.4)
3. Containment Water Level Monitor (II.F.1.5)
4. Containment Hydrogen Monitor (II.F.1.6)

A Notice of Consideration of Issuance of Amendments to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on May 21, 1985 (50 FR 20972). No public comments or requests for hearing were received.

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2.0 EVALUATION

1. Containment High-Range Radiation Monitor (II.F.1.3)

The licensee has installed two drywell radiation monitors in both Dresden Units that are consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-36 provided guidance for limiting conditions for operation and surveillance requirements for these monitors. The licensee has proposed TS that meet the intent of the guidance provided in Generic Letter 83-36. Therefore, the staff concludes that the proposed TS for Item II.F.1.3 are acceptable.

2. Containment Pressure Monitor (II.F.1.4)

Both Dresden Units have been provided with two wide range channels for monitoring drywell pressure following an accident. The licensee has proposed TS that contain limiting conditions for operation and surveillance requirements which meet the intent of the guidelines contained in Generic letter 83-36. Therefore, the staff concludes that the proposed TS for drywell pressure monitors are acceptable.

3. Torus Water Level Monitor (II.F.1.5)

The torus water level monitors at both Dresden Units provide the capability required by TMI Action Plan Item II.F.1.5. The TS for both units contain limiting conditions for operation and surveillance requirements that are consistent with the guidance contained in Generic Letter 83-36. Therefore, the staff concludes that the proposed TS for torus water level monitors are acceptable.

4. Containment Hydrogen Monitor (II.F.1.6)

The licensee installed drywell hydrogen monitors that provide the capability required by TMI Action Plan Item II.F.1.6. The proposed TS contain appropriate limiting conditions for operation and surveillance for these monitors. The staff concludes that the proposed TS are acceptable as they meet the intent of the guidance contained in Generic Letter 83-36.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. 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 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or 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 or to the health and safety of the public.

5.0 ACKNOWLEDGEMENT

This Safety Evaluation has been prepared by C. Patel.

Dated: June 24, 1985.