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November 10, 1986

Docket Nos. 50-237/249

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

Dear Mr. Farrar:

SUBJECT: MISCELLANEOUS CHANGES TO THE TECHNICAL SPECIFICATIONS (TACS 61209, 61210)

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

The Commission has issued the enclosed Amendment No. 94 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 90 to Facility Operating License No. DPR-25 for Dresden Unit 3. The amendments are in response to your application dated January 20, 1986 as supplemented by a letter dated July 29, 1986.

The amendments involve miscellaneous typographical errors, changes in nomenclature, sentence structure and references. They also include revisions to Table 3.7.1 to reflect modifications to Dresden 3 and operational changes to Dresden 2 to reflect Dresden Station compliance with the recommendations made in NUREG-0619 relating to Control Rod Drive nozzle cracking. Lastly, they modify Section 3.5.A.7 of the Dresden 3 Technical Specifications (TS) to allow the Core Spray and Low Pressure Coolant Injection systems to remain inoperable under specified conditions when the reactor is placed in the refuel mode from a cold shutdown condition. This change was previously found acceptable for Dresden 2 in Amendment No. 6 to Provisional Operating License No. DPR-19 issued April 16, 1975. The staff finds all of the above changes acceptable.

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

Sincerely,

Original signed by:

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 94 to License No. DPR-19
2. Amendment No. 90 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

DBL:BWD1	DBL:BWD1	OGC-BETH	DBL:BWD1
CJamerson	RGilbert		JZwolinski
10/30/86	10/30/86	~~10/~~ 5/86	11/10/86

Mr. Dennis L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:
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COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 94
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 January 20, 1986 as supplemented by a letter dated July 29, 1986 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 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. 94, 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, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 10, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 94

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
3/4.2-5	3/4.2-5
3/4.2-10	3/4.2-10
3/4.2-15	3/4.2-15
3/4.2-16	3/4.2-16
3/4.2-17	3/4.2-17
3/4.2-24	3/4.2-24
3/4.2-26	3/4.2-26
B3/4.2-30	B3/4.2-30
3/4.5-9	3/4.5-9
3/4.6-18	3/4.6-18
3/4.7-19	3/4.7-19
3/4.7-31	3/4.7-31
5-2	5-2

3.2 LIMITING CONDITION FOR OPERATION (CONT'D)

inoperability was not corrected in a timely manner. This is in lieu of an LER.

4. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

G. Radioactive Gaseous Effluent Instrumentation

1. The effluent monitoring instrumentation shown in Table 3.2.5 shall be operable with alarm/trip setpoints set to ensure that the limits of specification 3.8.A are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

2. With a radioactive gaseous effluent monitoring instrument alarm/trip setpoint less conservative

4.2 SURVEILLANCE REQUIREMENTS (CONT'D)

G. Radioactive Gaseous Effluent Instrumentation

Each radioactive gaseous radiation monitoring instrument in Table 4.2.3 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequency shown in Table 4.2.3.

TABLE 3.2.2
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Min. No. of Operable Inst. Channels per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	84" (plus 4, minus 0 inches) above top of active fuel (5)	1. In conjunction with low reactor pressure initiates core spray and LPCI. 2. In conjunction with high dry-well pressure, 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and SBGTS. 4. Initiates starting of diesel generators.
2	High Drywell Pressure (2), (3)	Less than or equal to 2 PSIG	1. Initiates core spray LPCI, HPCI, and SBGTS. 2. In conjunction with low low water level 120 sec. time delay and low pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators.
1	Reactor Low Pressure	Greater than or equal to 300 PSIG & less than or equal to 350 PSIG	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
1 (4)	Containment Spray Interlock 2/3 Core Height	Greater than or equal to 2/3 core height	Prevents inadvertent operation of containment spray during accident conditions.
2 (4)	Containment High Pressure	Greater than or equal to 0.5 PSIG & less than or equal to 1.5 PSIG	Prevents inadvertent operation of containment spray during accident conditions.
1	Timer Auto Blowdown	Less than or equal to 120 seconds	In conjunction with low low reactor water level, high dry-well pressure and low pressure core cooling interlock initiates auto blowdown.
2	Low Pressure Core Cooling Pump Discharge Pressure	Greater than or equal to 50 PSIG & less than or equal to 100 PSIG	* Defers APR actuation pending confirmation of low pressure core cooling system operation.
2/Bus	Under Voltage on 4 KV Emergency Buses	Greater than or equal to 3092 volts (Equals 3255 less) 5% tolerance)	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.
2	Sustained High Reactor Pressure	Less than or equal to 1070 PSIG for 15 seconds	Initiates isolation condenser.
2/Bus	Degraded Voltage on 4 KV Emergency Buses	Greater than or equal to 3708 volts (equals 3748 volts less 2% tolerance) after less than or equal to 5 minutes (plus 5% tolerance) with a 7 second (plus or minus 20%) inherent time delay	Initiates alarm and picks up time delay relay. Diesel generator picks up load if degraded voltage not corrected after time delay.

Notes: (See next page)

TABLE 3.2.5

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels (1)	Total No. of Channels	Parameter	Action (2)
1	2	Off-Gas Radiation Activity Monitor	D
1	3	Main Chimney Noble Gas SPING/GE Low Range Activity Monitor	A
1	1	Main Chimney SPING Noble Gas Monitors Mid, Hi Range	A
1	1	Main Chimney Iodine Sampler	C
1	1	Main Chimney Particulate Sampler	C
1	1	Main Chimney Flow Rate Monitor	B
1	1	Main Chimney Sampler Flow Rate Monitor	B
1	2	Reactor Building Vent Exhaust Duct Radiation Monitor	E
1	1	Reactor Building Vent SPING Noble Gas Monitor Low, Mid, High Range	F
1	1	Reactor Building Vent Flow Rate Monitor	B
1	1	Reactor Building Vent Sampler Flow Rate Monitor	B
1	1	Reactor Building Vent Iodine Sampler	C
1	1	Reactor Building Vent Particulate Sampler	C
1	1	MVRS Process Exhaust Iodine Sampler	E
1	1	MVRS Process Exhaust Particulate Sampler	E
1	1	MVRS HVAC Exhaust Iodine Sampler	E
1	1	MVRS HVAC Exhaust Particulate Sampler	E

Notes:
(See Next Page)

TABLE 3.2.5 (Notes)

1. For Off-Gas Radiation Monitors, applicable during SJAE operation. For other instrumentation, applicable at all times.

2. Action A: With the number of operable channels less than the minimum requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hour shift and these samples are analyzed within 24 hours.

   Action B: With the number of operable channels less than the minimum required, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per 4 hours.

   Action C: With less than the minimum channels operable, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment, as required in Table 4.8.1.

   Action D: With less than the minimum channels operable, gases from the main condenser off gas system may be released to the environment for up to 72 hours provided the off gas system is not bypassed and at least one chimney monitor is operable; otherwise, be in hot standby in 12 hours.

   Action E: With less than the minimum channels operable, immediately suspend release of radioactive effluents via this pathway.

   Action F: With less than the minimum channels operable, effluent releases via this pathway may continue provided that the minimum number of operable channels or the Reactor Building Vent Exhaust Duct Radiation Monitor are operable.

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	902-37	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	$0.1\text{-}10^6$ 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)

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)
Off-Gas 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
MVRS Process Exhaust Iodine and Particulate Sampler	D (7)	N/A	N/A	N/A
MVRS HVAC Exhaust Iodine and Particulate Sampler	D (7)	N/A	N/A	N/A

Notes: (See Next Page)

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	902-37	Once Every 12 Months	Once Per Day
	Torus Water Level Indicator (Narrow Range)	902-3	Once Every 6 Months	Once Per Day
2	(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)

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 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times 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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

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

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

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

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.) (Cont'd.)

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

4.6 SURVEILLANCE REQUIREMENT

3. When a snubber is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

B. Standby Gas Treatment System

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

   a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Standby Gas Treatment System

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

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperable for any reason and daily thereafter for the next succeeding seven days, initiate from the control room 4000 cfm (plus or minus 10%) flow through the operable circuit of the standby gas treatment system for at least 10 hours

TABLE 3.7.1
PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Listing of Power Operated Valves by Valve Number Outboard	Inboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation	(4)203-2A,B,C,D	(4)203-1A,B,C,D	3*T*5	0	GC
1	Main Steam Line Drain	220-2	220-1	* 35	C	SC
1	Recirculation Loop Sample (See Note 1)	220-45	220-44	* 5	C	SC
1	Isolation Condenser Vent	1301-20	1301-17	* 5	0	GC
2	Reactor Head Cooling	•	205-2-4	* 15	C	SC
2	Drywell Floor Drain	2001-106	2001-105	* 20	C	SC
2	Drywell Equipment Drain	2001-6	2001-5	* 20	C	SC
2	Drywell Vents	1601-23	1601-24	* 10	C	SC
2	Drywell Vent Relief		1601-62	* 15	C	SC
2	Drywell Inert & Purge		1601-21	* 10	C	SC
2	Drywell N_2 Makeup		1601-59	* 15	0	GC
2	Drywell/Torus N_2 Makeup	1601-57		* 15	0	GC
2	Drywell/Torus Inert	1601-55		* 15	0	GC
2	Torus N_2 Makeup		1601-58	* 15	C	SC
2	Torus Inert & Purge		1601-56	* 10	0	GC
2	Drywell & Torus Vent from Reactor Building	1601-22		* 10	C	SC
2	Drywell Vent to Standby Gas Treatment	1601-63		* 10	C	SC
2	Torus Vent		1601-60	* 10	C	SC
2	Torus Vent Relief		1601-61	* 15	C	SC
2	Drywell Air Sampling System (See Note 1)	(7)9205A, 9206A, 9207B, 9208B, 8501-1B, 8501-3B, 8501-5B	(7)9205B, 9206B, 9207A, 9208A, 8501-1A, 8501-3A, 8501-5A	* 5	0	GC
2	Torus to Condenser Drain	1599-62	1599-61	* 10	C	SC
2	Drywell Pneumatic Supply	4721	4720	* 10	0	GC
3	Cleanup Demineralizer system	1201-2	1201-1	* 30	0	GC
3	Cleanup Demineralizer System	1201-3	1201-1A	* 30	C	SC
3	Shutdown Cooling	(3)1001-2A,B,C	(4)1001-1A, 1B 1001-5A,B	* 40	C	SC
4	HPCI Turbine Steam Supply	2301-4	2301-5	* 25	0	GC
4	HPCI Torus Suction	2301-35	2301-36	* 80	C	SC
5	Isolation Condenser Steam Supply	1301-2	1301-1	* 30	0	GC
5	Isolation Condenser Condensate Return		1301-4	* 30	0	GC
5	Isolation Condenser Condensate Return	1301-3		* 30	C	SC
N/A	Feedwater Check Valves	220-62A,62B	220-58A,58B	N/A	0	Process
N/A	Control Rod Hydraulic Return Check Valves	301-95	301-98	N/A	C	Process
N/A	Reactor Head Cooling Check Valves		205-2-7	N/A	C	Process
N/A	Standby Liquid Control Check Valves	1101-16	1101-15	N/A	C	Process
N/A	Core Spray Injection	(2)1401-24A,24B		N/A	0	N/A
			(2)1402-25A,25B	N/A	C	N/A
N/A	Core Spray Test Return		(2)1402-4A,4B	N/A	C	N/A
N/A	Core Spray Suction		(2)1402-3A,3B	N/A	0	N/A
N/A	LPCI Torus Spray	(2)1501-18A,18B	(2)1501-19A,19B	N/A	C	N/A
N/A	LPCI Test Return	(2)1501-20A,20B	(2)1501-38A,38B	N/A	C	N/A
N/A	LPCI Injection	(2)1501-22A,22B	(2)1501-25A,25B	N/A	C	N/A
N/A	LPCI Drywell Spray	(2)1501-27A,27B	(2)1501-28A,28B	N/A	C	N/A
N/A	LPCI Suction		(4)1501-5A,5B,5C,5D	N/A	0	N/A

Notes: (See Next Page)

DESIGN FEATURES (Cont'd.)

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 20 per cent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

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. 90
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 January 20, 1986 as supplemented by a letter dated July 29, 1986 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. 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, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 10, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 90

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
3/4.2-5	3/4.2-5
3/4.2-10	3/2.2-10
3/4.2-15	3/4.2-15
3/4.2-16	3/4.2-16
3/4.2-17	3/4.2-17
3/4.2-24	3/4.2-24
3/4.2-26	3/4.2-26
B3/4.2-30	B3/4.2-30
3/4.5-4	3/4.5-4
3/4.5-9	3/4.5-9
3/4.7-19	3/4.7-19
3/4.7-31	3/4.7-31
B3/4.7-40	B3/4.7-40
5-2	5-2

3.2 LIMITING CONDITION FOR OPERATION (CONT'D)

inoperability was not corrected in a timely manner. This is in lieu of an LER.

4. In the event a limiting condition for operation and associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specifications, provide a 30-day written report to the NRC pursuant to Specification 6.6.B.2., and no changes are required in the operational condition of the plant, and this does not prevent the plant from entry into an operational mode.

G. Radioactive Gaseous Effluent Instrumentation

1. The effluent monitoring instrumentation shown in Table 3.2.5 shall be operable with alarm/trip setpoints set to ensure that the limits of specification 3.8.A are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

2. With a radioactive gaseous effluent monitoring instrument alarm/trip setpoint less conservative

4.2 SURVEILLANCE REQUIREMENTS (CONT'D)

G. Radioactive Gaseous Effluent Instrumentation

Each radioactive gaseous radiation monitoring instrument in Table 4.2.3 shall be demonstrated operable by performance of the given source check, instrument check, calibration, and functional test operations at the frequency shown in Table 4.2.3.

TABLE 3.2.2
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Min. No. of Operable Inst. Channels per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	Reactor Low Low Water Level	84" (plus 4, minus 0 inches) above top of active fuel (5)	1. In conjunction with low reactor pressure initiates core spray and LPCI. 2. In conjunction with high dry-well pressure, 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates HPCI and SBGTS. 4. Initiates starting of diesel generators.
2	High Drywell Pressure (2), (3)	Less than or equal to 2 PSIG	1. Initiates core spray, LPCI, HPCI, and SBGTS. 2. In conjunction with low low water level 120 sec. time delay, and low pressure core cooling interlock initiates auto blowdown. 3. Initiates starting of diesel generators.
1	Reactor Low Pressure	Greater than or equal to 300 PSIG & less than or equal to 350 PSIG	1. Permissive for opening core spray and LPCI admission valves. 2. In conjunction with low low reactor water level initiates core spray and LPCI.
1 (4)	Containment Spray Inter-lock 2/3 Core Height	Greater than or equal to 2/3 core height	Prevents inadvertent operation of containment spray during accident conditions.
2 (4)	Containment High Pressure	Greater than or equal to 0.5 PSIG & less than or equal to 1.5 PSIG	Prevents inadvertent operation of containment spray during accident conditions.
1	Timer Auto Blowdown	Less than or equal to 120 seconds	In conjunction with low low reactor water level, high dry-well pressure, and low pressure core cooling interlock initiates auto blowdown.
2	Low Pressure Core Cooling Pump Discharge Pressure	Greater than or equal to 50 PSIG & less than or equal to 100 PSIG.	* Defers APR actuation pending confirmation of low pressure core cooling system operation.
2/Bus	Under Voltage on 4 KV Emergency Buses	Greater than or equal to 3092 volts (Equals 3255 less 5% tolerance)	1. Initiates starting of diesel generators. 2. Permissive for starting ECCS pumps. 3. Removes nonessential loads from buses.
2	Sustained High Reactor Pressure	Less than or equal to 1070 PSIG for 15 seconds	Initiates isolation condenser.
2/Bus	Degraded Voltage on 4 KV Emergency Buses	Greater than or equal to 3708 volts (equals 3748 volts less 2% tolerance) after less than or equal to 5 minutes (plus 5% tolerance) with a 7 second (plus or minus 20%) inherent time delay	Initiates alarm and picks up time delay relay. Diesel generator picks up load if degraded voltage not corrected after time delay.

Notes: (See next page)

TABLE 3.2.5
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Minimum No. of Operable Channels (1)	Total No. of Channels	Parameter	Action (2)
1	2	Off-Gas Radiation Activity Monitor	D
1	3	Main Chimney Noble Gas SPING/GE Low Range Activity Monitor	A
1	1	Main Chimney SPING Noble Gas Monitors Mid, Hi Range	A
1	1	Main Chimney Iodine Sampler	C
1	1	Main Chimney Particulate Sampler	C
1	1	Main Chimney Flow Rate Monitor	B
1	1	Main Chimney Sampler Flow Rate Monitor	B
1	2	Reactor Building Vent Exhaust Duct Radiation Monitor	E
1	1	Reactor Building Vent SPING Noble Gas Monitor Low, Mid, High Range	F
1	1	Reactor Building Vent Flow Rate Monitor	B
1	1	Reactor Building Vent Sampler Flow Rate Monitor	B
1	1	Reactor Building Vent Iodine Sampler	C
1	1	Reactor Building Vent Particulate Sampler	C
1	1	MVRS Process Exhaust Iodine Sampler	E
1	1	MVRS Process Exhaust Particulate Sampler	E
1	1	MVRS HVAC Exhaust Iodine Sampler	E
1	1	MVRS HVAC Exhaust Particulate Sampler	E

Notes: -
(See Next Page)

TABLE 3.2.5 (Notes)

1. For Off-Gas Radiation Monitors, applicable during SJAE operation. For other instrumentation, applicable at all times.

2. Action A: With the number of operable channels less than the minimum requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hour shift and these samples are analyzed within 24 hours.

Action B: With the number of operable channels less than the minimum required, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per 4 hours.

Action C: With less than the minimum channels operable, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment, as required in Table 4.8.1.

Action D: With less than the minimum channels operable, gases from the main condenser off gas system may be released to the environment for up to 72 hours provided the off gas system is not bypassed and at least one chimney monitor is operable; otherwise, be in hot standby in 12 hours.

Action E: With less than the minimum channels operable, immediately suspend release of radioactive effluents via this pathway.

Action F: With less than the minimum channels operable, effluent releases via this pathway may continue provided that the minimum number of operable channels or the Reactor Building Vent Exhaust Duct Radiation Monitor are operable.

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	903-37	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)

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)
Off-Gas 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
MVRS Process Exhaust Iodine and Particulate Sampler	D(7)	N/A	N/A	N/A
MVRS HVAC Exhaust Iodine and Particulate Sampler	D(7)	N/A	N/A	N/A

Notes:
(See Next Page)

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	903-37	Once Every 12 Months	Once Per Day
	Torus Water Level Indicator (Narrow Range)	903-3	Once Every 6 Months	Once Per Day
2	(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 Main Steam RV Position, Acoustic Monitor	903-21	Once Every Refuel (1)	Once Per 31 Days
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)

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 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times 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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

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

7. If the requirements of 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours. Subsequently, the reactor may be placed in Refuel, for post maintenance testing of control rod drives only, provided no work is being performed which has the potential to drain the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

the containment cooling subsystem, shall be demonstrated to be operable immediately and daily thereafter.

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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the HPCI Subsystem is operable.

3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.

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

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately.

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

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

B. Standby Gas Treatment System

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

   a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

B. Standby Gas Treatment System

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

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperable for any reason and daily thereafter for the next succeeding seven days, initiate from the control room 4000 cfm (plus or minus 10%) flow through the operable circuit of the standby gas treatment system with the circuit

TABLE 3.7.1
PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Listing of Power Operated Valves by Valve Number: Outboard	Listing of Power Operated Valves by Valve Number: Inboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation	(4)203-2A,B,C,D	(4)203-1A,B,C,D	3*T*5	O	GC
1	Main Steam Line Drain	220-2	220-1	* 35	C	SC
1	Recirculation Loop Sample (See Note 1)	220-45	220-44	* 5	C	SC
1	Isolation Condenser Vent	1301-20	1301-17	* 5	O	GC
2	Reactor Head Cooling		205-2-4	* 15	C	SC
2	Drywell Floor Drain	2001-106	2001-105	* 20	C	SC
2	Drywell Equipment Drain	2001-6	2001-5	* 20	C	SC
2	Drywell Vents	1601-23	1601-24	* 10	C	SC
2	Drywell Vent Relief		1601-62	* 15	C	SC
2	Drywell Inert & Purge		1601-21	* 10	C	SC
2	Drywell N_2 Makeup		1601-59	* 15	O	GC
2	Drywell/Torus N_2 Makeup	1601-57		* 15	O	GC
2	Drywell/Torus Inert	1601-55		* 15	O	GC
2	Torus N_2 Makeup		1601-58	* 15	C	SC
2	Torus Inert & Purge		1601-56	* 10	O	GC
2	Drywell & Torus Vent from Reactor Building	1601-22		* 10	C	SC
2	Drywell Vent to Standby Gas Treatment	1601-63		* 10	C	SC
2	Torus Vent		1601-60	* 10	C	SC
2	Torus Vent Relief		1601-61	* 15	C	SC
2	Drywell Air Sampling System (See Note 1)	(7)9205A, 9206A, 9207B, 9208B, 8501-1B, 8501-3B, 8501-5B	(7)9205B, 9206B, 9207A, 9208A, 8501-1A, 8501-3A, 8501-5A	* 5	O	GC
2	Torus to Condenser Drain	1599-62	1599-61	* 10	C	SC
2	Drywell Pneumatic Supply	4721	4720	* 10	O	GC
3	Cleanup Demineralizer system	1201-2	1201-1	* 30	O	GC
3	Cleanup Demineralizer System	1201-3	1201-1A	* 30	C	SC
3	Shutdown Cooling	(3)1001-2A,B,C	(4)1001-1A, 1B 1001-5A,B	* 40	C	SC
4	HPCI Turbine Steam Supply	2301-4	2301-5	* 25	O	GC
4	HPCI Torus Suction	2301-35	2301-36	* 80	C	SC
5	Isolation Condenser Steam Supply	1301-2	1301-1	* 30	O	GC
5	Isolation Condenser Condensate Return		1301-4	* 30	O	GC
5	Isolation Condenser Condensate Return	1301-3		* 30	C	SC
N/A	Feedwater Check Valves	220-62A,62B	220-58A,58B	N/A	O	Process
N/A	Reactor Head Cooling Check Valves		205-2-7	N/A	C	Process
N/A	Standby Liquid Control Check Valves	1101-16	1101-15	N/A	C	Process
N/A	Core Spray Injection	(2)1401-24A,24B		N/A	O	N/A
			(2)1402-25A,25B	N/A	C	N/A
N/A	Core Spray Test Return		(2)1402-4A,4B	N/A	C	N/A
N/A	Core Spray Suction		(2)1402-3A,3B	N/A	O	N/A
N/A	LPCI Torus Spray	(2)1501-18A,18B	(2)1501-19A,19B	N/A	C	N/A
N/A	LPCI Test Return	(2)1501-20A,20B	(2)1501-38A,38B	N/A	C	N/A
N/A	LPCI Injection	(2)1501-22A,22B	(2)1501-25A,25B	N/A	C	N/A
N/A	LPCI Drywell Spray	(2)1501-27A,27B	(2)1501-28A,28B	N/A	C	N/A
N/A	LPCI Suction		(4)1501-5A,5B,5C,5D	N/A	O	N/A

Notes: (See Next Page)

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

standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

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

4.7 SURVEILLANCE REQUIREMENT BASES

A. Primary Containment

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

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

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3123A

DESIGN FEATURES (Cont'd)

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 20 per cent of gravity. Dynamic analysis was used to determine the earthquake acceleration, applicable to the various elevations in the reactor building.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 90 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

By a letter dated January 20, 1986 as supplemented by a letter dated July 29, 1986, Commonwealth Edison Company (the licensee) proposed to amend Appendix A of Provisional Operating License (POL) No. DPR-19 and Facility Operating License No. DPR-25. The letters provided information to support changes which were primarily to correct typographical errors, changes in nomenclature, sentence structure and references to improve Technical Specification (TS) clarity with the exception of a change for Dresden Unit 3 to allow post-maintenance testing of control rod drives in the refuel mode with low pressure cooling systems inoperable. This change was approved on April 16, 1975 for Dresden 2 in Amendment 6 to POL No. DPR-19. The July 29, 1986 letter also revised Table 3.7.1 of Appendix A of both licenses to reflect the results of minor appropriate plant modifications recently implemented to bring the units into compliance with staff recommendations in NUREG-0619, "BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking."

2.0 EVALUATION

The staff has carefully examined the TS pages submitted by the licensee which reflect corrections to typographical errors, changes in nomenclature, sentence structure and references and finds that the changes improve the clarity of the TS without changing the technical content. They are, therefore, acceptable.

The licensee attached to the January 20, 1986 letter a copy of the April 16, 1975 amendment package allowing Dresden Unit 2 to perform post-maintenance testing of control rod drives in the refuel mode (following achievement of cold shutdown) with low pressure cooling systems inoperable provided that no work is being done which has the potential for draining the reactor vessel. The staff reviewed this package and finds it supports the licensee's request for an identical change for Dresden Unit 3 and that the revisions to Dresden Unit 3's TS documenting this are acceptable.

During the recent Dresden Unit 3 recirculation pipe replacement outage, the Control Rod Drive (CRD) return line to the reactor vessel was permanently removed from inside containment and the containment penetration was capped. Outside containment, the pipe was cut and the necessary caps installed. This deleted CRD valve 3-0301-98 and isolated CRD valve 3-0301-95 from primary containment. Since these valves no longer serve as primary containment isolation valves for Unit 3, they are being removed from Dresden 3 TS Table 3.7.1. The staff finds this acceptable.

The CRD return line for Dresden Unit 2 is always valved out via the two CRD return check valves 2-0301-95 and 2-0301-98. Table 3.7.1 of the Dresden Unit 2 TS is being changed to reflect their normal position as closed. The staff finds that this meets the recommendations in NUREG-0619 and is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 CONCLUSION

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

Principal Contributor: R. Gilbert

Dated: November 10, 1986