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

November 3, 1988

Docket Nos.: 50-237
and 50-249

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT TO REFLECT INSTRUMENTATION CHANGES RESULTING FROM POST-ACCIDENT MONITORING CHANGES COMPLETED PER REGULATORY GUIDE 1.97 AND NUREG 0737, SUPPLEMENT 1 (TAC NOS. 68720 and 68721)

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

The Commission has issued the enclosed Amendment No. 102 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 98 to Facility Operating License No. DPR-25 for Dresden Unit 3. The amendments are in response to your application dated June 20, 1988.

The amendments make changes to the Technical Specifications for Dresden Units 2 and 3 to reflect instrumentation enhancements for post-accident monitoring completed per Regulatory Guide 1.97 and NUREG-0737 Supplement 1. In addition several minor corrections and clarifications have been incorporated.

On July 25, 1988 CECo notified the staff that the primary oxygen monitor that meets Regulatory Guide 1.97 had been disconnected and inoperable for an extended period of time. Although the staff position has been that instrumentation classified as Category 1 in Regulatory Guide 1.97 should be included in the Standard Technical Specifications this Category 1 oxygen monitor is not included in the Dresden Technical Specifications. On the basis of this recent occurrence at Dresden, which identified that the oxygen monitoring instrumentation identified in Regulatory Guide 1.97 as Category 1 has not been included in your Technical Specifications, the staff recommends

DF01
1/1

that CECo re-examine the applicable sections of the Dresden Technical Specifications to assure that they are consistent with guidance contained in Regulatory Guide 1.97. A proposed amendment should be submitted to correct any omissions identified.

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

Sincerely,

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

Enclosures:
1. Amendment No. 102 to License No. DPR-19
2. Amendment No. 98 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

~~Docket file~~	PDIII-2 Plant file
NRC & Local PDRs	GPA/PA
PDIII-2 r/f	ACRS (10)
GHolahan	EJordan
LLuther	OCG-Rockville
BSiegel	MRing, III
TBarnhart (8)	ARM/LFMB
DHagan	BGrimes
WJones	EButcher
MVirgilio	BMarcus
JJoyce	

*See previous concurrences

PDIII-2	PDIII-2	PDIII-2	*OGC
*BSiegel:km	*LLuther	*DMuller	
/ /88	/ /88	/ /88	/ /88

0F01
1/1

recommends that CECo re-examine the applicable sections of the Dresden Technical Specifications to assure that they are consistent with guidance contained in Regulatory Guide 1.97 and the Murley May 9, 1988 letter to the BWR Owners Group. A proposed amendment should be submitted to correct any omissions identified in addition to the oxygen monitor.

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

Sincerely,

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

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

cc w/enclosures:
See next page

DISTRIBUTION

Docket file	PDIII-2 Plant file
NRC & Local PDRs	GPA/PA
PDIII-2 r/f	ACRS (10)
GHolahan	EJordan
LLuther	OCG-Rockville
BSiegel	MRing, III
TBarnhart (8)	ARM/LFMB
DHagan	BGrimes
WJones	EButcher
MVirgilio	BMarcus
JJoyce	

PDIII-2	PDIII-2	PDIII	OGC
BSiegel:km	LLuther	DMuller	RBachmann
09/06/88	9/06/88	9/6/88	9/12/88

that CECo re-examine the applicable sections of the Dresden Technical Specifications to assure that they are consistent with guidance contained in Regulatory Guide 1.97. A proposed amendment should be submitted to correct any omissions identified.

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

Sincerely,

Byron Siegel

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

Enclosures:
1. Amendment No. 102 to
   License No. DPR-19
2. Amendment No. 98 to
   License No. DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Henry E. Bliss
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:

Michael I. Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Mr. J. Eenigenburg
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
Rural Route #1
Morris, Illinois 60450

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Mr. Michael E. Parker, Chief
Division of Engineering
Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

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. 102
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 June 20, 1988 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. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 102

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

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

REMOVE	INSERT
3/4.2-17 (Table 3.2.6)	3/4.2-17 (Table 3.2.6)
3/4.2-19 (Table 4.2.1)	3/4.2-19 (Table 4.2.1)
3/4.2-26 (Table 4.2.4)	3/4.2-26 (Table 4.2.4)
B 3/4.2-36	B 3/4.2-36
B 3/4.2-37	B 3/4.2-37
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3

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	0-1500 psig
			2	0-1200 psig
		902-3	1	0-1500 psig
1	Reactor Water Level	902-3,5	2	-340 to +60 inches
1	Torus Bulk Water Temperature	902-4,37	2	0-300°F
2 (3)	Torus Water Level Indicator	902-3	1	-20 to +20 inches (narrow range)
		902-2	2	0-30 ft (wide range)
	Torus Water Local Sight Glass		1	40 inch range (narrow range)
1 (4)	Torus Pressure	902-5	1	-2.45 to +5 psig
2	Drywell Pressure	902-5	1	0-5 psig
		902-3	1	-5 to +70 psig
		902-3	2	-5 to +250 psig
2	Drywell Temperature	902-3	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
	Main Steam RV Position,			
1/valve (2)	Acoustic Monitor	902-21	1 per valve	N/A
1/valve (2)	Temperature Monitor	902-21	1 per valve	0-600°F
	Main Steam SV Position,			
1/valve (2)	Acoustic Monitor	902-21	1 per valve	N/A
1/valve (2)	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 4.2.1

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. -5 to +70 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. -20 to +20 inches Narrow Range Indicator			
b. 40 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)

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	902-3,5	Once Every 6 Months	Once Per Day
1	Reactor Water Level	902-3,5	Once Every 6 Months	Once Per Day
1	Torus Bulk Water Temperature	902-4,37	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
	(-5 to +70 psig)	902-3	Once Every 3 Months	Once Per 31 Days
	(-5 to 250 psig)	902-3	Once Every Refuel	Once Per 31 Days
2	Drywell Temperature	902-3	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
1/Valve	Main Steam RV Position, Temperature Monitor	902-21	Once Every Refuel	Once Per 31 Days
1/Valve	Acoustic Monitor	902-21	(1)	Once Per 31 Days
1/Valve	Main Steam SV Position, Temperature Monitor	902-21	Once Every Refuel	Once Per 31 Days
1/Valve	Acoustic Monitor	902-21	(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)

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, 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 2-2389A, 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 2-263-73A, 73B and 2-2352, 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.

3.7 LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by Specification 3.5.F.3, 3.5.F.4, the suppression pool water volume and bulk water temperature shall be maintained within the following limits.

   a. Maximum water volume - 115,655 ft^3

   b. Minimum water volume - 112,000 ft^3

4.7 SURVEILLANCE REQUIREMENTS

CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The surveillances are as follows:

   a. The suppression pool water level and bulk water temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Maximum bulk water temperature

(1) During normal power operation; maximum 95°F bulk water temperature.

(2) During testing which adds heat to the suppression pool, the bulk water temperature shall not exceed 10° F above the normal power operation limit specified in (1) above. In connection with such testing, the pool bulk water temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

testing which adds heat to the suppression pool, the pool bulk water temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the bulk water temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 150 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The reactor shall be scrammed from any operating condition if the pool bulk water temperature reaches 110°F.

Power operation shall not be resumed until the pool bulk water temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool bulk water temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

e. Minimum downcomer submergence is 3.67 ft.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

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. 98
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 June 20, 1988 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. 98, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 98

FACILITY OPERATING LICENSE DPR-25

DOCKET NO. 50-249

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

REMOVE	INSERT
3/4.2-17 (Table 3.2.6)	3/4.2-17 (Table 3.2.6)
3/4.2-19 (Table 4.2.1)	3/4.2-19 (Table 4.2.1)
3/4.2-26 (Table 4.2.4)	3/4.2-26 (Table 4.2.4)
3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3

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	903-3 903-3,5	1 2	0-1500 psig -340 to +60 inches
1	Torus Bulk Water Temperature	903-4,37	2	0-300°F
2 (3)	Torus Water Level Indicator	903-3	1	-20 to +20 inches (narrow range)
		903-2	2	0-30 ft (wide range)
	Torus Water Local Sight Glass		1	40 inch range (narrow range)
1 (4)	Torus Pressure	903-5	1	-2.45 to +5 psig
2	Drywell Pressure	903-5 903-3 903-3	1 1 2	0-5 psig -5 to +70 psig -5 to +250 psig
2	Drywell Temperature	903-3	6	0-600°F
2	Neutron Monitoring	903-5	4	0.1-16^6 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
	Main Steam RV Position,			
1/valve (2)	Acoustic Monitor	903-21	1 per valve	N/A
1/valve (2)	Temperature Monitor	903-21	1 per valve	0-600°F
	Main Steam SV Position,			
1/valve (2)	Acoustic Monitor	903-21	1 per valve	N/A
1/valve (2)	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 4.2.1

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)	Refuel Outage	None
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) (3)	(2) (3)	(2)
	(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. -5 to +70 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. -20 to +20 inches Narrow Range Indicator			
b. 40 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)

TABLE 4.2.4

Post Accident Monitoring Instrumentation Surveillance Requirements

Minimum Number of Operable Channels (1)	Parameter	Instrument Readout Location Unit 3	Calibration	Instrument Check
1	Reactor Pressure	903-3,5	Once Every 6 Months	Once Per Day
1	Reactor Water Level	903-3,5	Once Every 6 Months	Once Per Day
1	Torus Bulk Water Temperature	903-4,37	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)	903-2	N/A	None
	(Wide Range)		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
	(-5 to +70 psig)	903-3	Once Every 3 Months	Once Per 31 Days
	(-5 to +250 psig)	903-3	Once Every Refuel	Once Per 31 Days
2	Drywell Temperature	903-3	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
	Main Steam RV Position			
1/Valve	Temperature Monitor	903-21	Once Every Refuel (1)	Once Per 31 Days
1/Valve	Acoustic Monitor	903-21		Once Per 31 Days
	Main Steam SV Position			
1/Valve	Temperature Monitor	903-21	Once Every Refuel (1)	Once Per 31 Days
1/Valve	Acoustic Monitor	903-21		Once Per 31 Days
1	Drywell Hydrogen Concentration	903-55 903-56	Once Every 3 Months	Once Per 31 Days

3.7 LIMITING CONDITION FOR OPERATION

CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, except as permitted by Specification 3.5.F.3, 3.5.F.4, or 3.5.F.5, the suppression pool water volume and bulk water temperature shall be maintained within the following limits.

   a. Maximum water volume - 115,655 ft^3

   b. Minimum water volume - 112,000 ft^3

4.7 SURVEILLANCE REQUIREMENTS

CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. The surveillances are as follows:

   a. The suppression pool water level and bulk water temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

c. Maximum bulk water temperature

(1) During normal power operation; maximum 95°F bulk water temperature.

(2) During testing which adds heat to the suppression pool, the bulk water temperature shall not exceed 10° F above the normal power operation limit specified in (1) above. In connection with such testing, the pool bulk water temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

testing which adds heat to the suppression pool, the pool bulk water temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the bulk water temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 150 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(3) The reactor shall be scrammed from any operating condition if the pool bulk water temperature reaches 110°F.

Power operation shall not be resumed until the pool bulk water temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool bulk water temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

e. Minimum downcomer submergence is 3.67 ft.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 102 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

Commonwealth Edison (the licensee or CECo) was requested by Generic Letter 82-33 (Supplement 1 to NUREG-0737) to provide a report to the NRC describing how the post-accident monitoring instrumentation meets the guidelines of Regulatory Guide 1.97 as applied to the emergency response facilities. The licensee provided a response to the generic letter on August 1, 1985. Additional information was provided by letters dated November 4, 1985, January 31, 1986, July 1, 1986 and October 28, 1987.

A detailed review and technical evaluation of the CECo's submittals was performed by EG&G Idaho, Inc. under contract to the NRC with general supervision from the NRC staff. The staff's Safety Evaluation of this issue was transmitted to CECo by letter dated September 1, 1988.

In a letter dated June 20, 1988 CECo submitted proposed Technical Specification changes for Dresden Unit Nos. 2 and 3 to reflect instrumentation enhancements for post-accident monitoring completed per Regulatory Guide 1.97 and NUREG-0737 Supplement 1 or resulting from the staff's Detailed Control Room Design Review (DCRDR). The proposed revisions to the Technical Specifications involve changing instrument ranges and panel locations associated with drywell temperature and pressure, reactor level, and torus water level and temperature instrumentation. In addition several administrative changes in the form of correction of typographical errors and clarifications were also included which will not be discussed further.

2.0 EVALUATION

Eight drywell atmospheric monitoring thermocouples have been replaced by environmentally qualified thermocouples and cable from the drywell to the control room. The old thermocouples went to recorder on a back panel. Two of the new thermocouples go the the main computer and input to the safety parameter display system. The remaining six thermocouples go to a recorder located on a front panel. These modifications were made in accordance with CECo's commitments to meet the post-accident monitoring guidelines of Regulatory Guide 1.97 and HED's identified during the DCRDR that required a temperature recorder to be located on a front panel to monitor the drywell atmosphere temperature. The staff has concluded that CECo's proposed amendment request corrects the applicable Technical Specifications, with regard to the changed instrument locations, to satisfy the Regulatory Guide 1.97 commitments and correct the HED identified and is therefore acceptable.

The licensee has stated that the midrange drywell pressure indicator will be recalibrated from 0 to 75 psig to -5 to +70 psig. The licensee identified this as an exception to Regulatory Guide 1.97 in its responses to Generic Letter 82-33. This exception from the conformance to Regulatory Guide 1.97 was evaluated and found acceptable by the staff in its Safety Evaluation transmitted to the licensee in a letter dated September 1, 1988 (Section 3.3.2 of contractor Technical Evaluation Report attached to staff's Safety Evaluation). The proposed Technical Specification change is therefore acceptable.

The licensee has also made the following instrumentation enhancements to conform with the recommendations of Regulatory Guide 1.97: placement of a redundant wide range reactor vessel pressure indicator on panel 902(3)-3; placement of wide range reactor vessel level pen indicator on the 902(3)-5 continuous recorder; changing of the torus water level sight glass range; and changing of the torus water temperature range. The staff has evaluated the proposed Technical Specification changes associated with the instrumentation enhancements described above and determined that since these enhancements have been made to comply with the guidance contained in Regulatory Guide 1.97 they are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to a requirement with respect to the installation or use of a facility component 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: Byron L. Siegel

Dated: November 3, 1988