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BJones (8)
BScharf (10)
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OPA/Clare Miles
TBAbernathy
JRBuchanan
ACRS (16)
RPSnaider

SEP 29 1976

Docket Nos. 50-237
50-249

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your requests dated December 10, 1974, October 1, 1975, and March 17, 1976, the Commission has issued the enclosed Amendment Nos. 25 and 22 to Facility Operating License Nos. DPR-19 and DPR-25 for Unit Nos. 2 and 3 of the Dresden Nuclear Power Station, respectively.

The amendments consist of changes in the Technical Specifications that add new sections 3.6.I and 4.6.I which identify safety related shock suppressors and include requirements regarding operability and surveillance of these shock suppressors (snubbers). Some minor modifications to the proposed Technical Specifications have been made as discussed with your staff.

A copy of the related Safety Evaluation and Notice of Issuance also are enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 25 to License No. DPR-19
2. Amendment No. 22 to License No. DPR-25
3. Safety Evaluation
4. Notice

Amdts 25 + 22

cc w/enclosures:
See next page

OFFICE ➤	OR:ORB #2	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2	
SURNAME ➤	RMDiggs	RDSilver	RPSnaider:ro	DSwanson	DLZiemann	
DATE ➤	9/9/76	9/17/76	9/14/76	9/27/76	9/29/76	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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of Grundy County
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filings dtd. 12/10/74, 10/1/75
and 3/17/76:
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Bureau of Radiological Health
Illinois Department of Public Health
Springfield, Illinois 62706

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. 25
License No. 19

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated December 10, 1974, October 1, 1975, and March 17, 1976, comply 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.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following existing pages of the Technical Specifications with the attached revised pages bearing the same numbers, except as otherwise indicated. Changed areas on the revised pages are shown by marginal lines.

REMOVE	INSERT
Pages iii of Table of Contents	Pages iii of Table of Contents
-	91b (new page)
-	91c (new page)
-	91d (new page)
-	91e (new page)
99a	99a
-	99b (new page)

TABLE OF CONTENTS (Cont)

3.6 LIMITING CONDITION FOR OPERATION

I. Shock Suppressors (Snubbers)

1. During all modes of operation except cold shutdown and refuel, all safety-related snubbers listed in Table 3.6.1 shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.I.1 and 3.6.I.2 can not 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.

5. Snubbers may be added to safety related systems without prior license amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next license amendment request.

4.6 SURVEILLANCE REQUIREMENT

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Table 3.6.1.

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability in accordance with the following schedule:

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. The initial inspection shall be performed within six months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.I.1, it shall be assumed that the facility had been on a six-month inspection interval.

4. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lbs. need not be functionally tested.

TABLE 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Azimuth	Snubber in High* Radiation Area During Shutdown	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
2	Torus Ring Header 1501-24"	483'	83°			X
3	Torus Ring Header 1501-24"	483'	74°			X
4	Torus Ring Header 1501-24"	483'	38°			X
5	Torus Ring Header 1501-24"	483'	29°			X
7	Torus Ring Header 1501-24"	483'	331°			X
8	Torus Ring Header 1501-24"	483'	322°			X
9	Torus Ring Header 1501-24"	483'	286°			X
10	Torus Ring Header 1501-24"	483'	277°			X
12	Torus Ring Header 1501-24"	483'	218°			X
13	Torus Ring Header 1501-24"	483'	209°			X
15	Torus Ring Header 1501-24"	483'	151°			X
16	Torus Ring Header 1501-24"	483'	142°			X
1	Drywell Recirc. Motor 2B-202	524'	328°	X	X	
2	Drywell Recirc. Motor 2B-202	524'	302°	X	X	
3	Drywell Recirc. Motor 2B-202	524'	315°	X	X	
4	Drywell Recirc. Motor 2A-202	524'	148°	X	X	
5	Drywell Recirc. Motor 2A-202	524'	122°	X	X	
6	Drywell Recirc. Motor 2A-202	524'	135°	X	X	
7	Drywell Recirc. Pump 2B-202	512'	326°	X	X	
8	Drywell Recirc. Pump 2B-202	512'	304°	X	X	
9	Drywell Recirc. Pump 2B-202	507'	315°	X	X	
10	Drywell Recirc. Pump 2A-202	512'	124°	X	X	
11	Drywell Recirc. Pump 2A-202	512'	146°	X	X	
12	Drywell Recirc. Pump 2A-202	507'	135°	X	X	
13	Drywell Recirc. Line 201B-28"	507'	305°	X	X	
14	Drywell Recirc. Line 201A-28"	507'	105°	X	X	

TABLE 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Azimuth	Snubber in High* Radiation Area During Shutdown	Snubbers Inaccessable During Normal Operation	Snubbers Accessible During Normal Operation
15	Drywell LPCI Line 1506-16"	513'	95°	X	X	
16	Drywell LPCI Line 1519-16"	513'	256°	X	X	
17	Drywell Recirc. Header 201B-22"	533'6"	195°	X	X	
18	Drywell HPCI Line 2305-10"	531'	117°	X	X	
19	Drywell HPCI Line 2305-10"	533'	112°	X	X	
20	Drywell HPCI Line 2305-10"	533'	108°	X	X	
21	Drywell Recirc. Header 201A-22"	533'6"	22°	X	X	
22	Drywell HPCI Line 2305-10"	550'	121°	X	X	
23	Drywell Cleanup Line 1201-8"	533'	330°	X	X	
24	Drywell Feedwater Line 3204D-12"	538'	106°	X	X	
25	Drywell Cleanup Line 1201-8"	534'	325°	X	X	
26	Drywell Feedwater Line 3204C-12"	540'	73°	X	X	
27	Drywell Cleanup 1201-8"	534'6"	300°	X	X	
28	Drywell Feedwater Line 3204A-18"	537'6"	66°	X	X	
29	Drywell HPCI Line 2305-10"	563'	140°	X	X	
30	Drywell Core Spray Line 1403-10"	575'	336°	X	X	
31	Drywell Core Spray Line 1404-10"	562'	231°	X	X	
32	Drywell Target Rock Valve 203-3A	542'6"	16°	X	X	
33	Drywell Target Rock Valve 203-3A	542'4"	31°	X	X	
34	Drywell Target Rock Valve 203-3A	540'0"	19°	X	X	
35	Drywell Target Rock Valve 203-3A	540'3"	34°	X	X	
	Isolation Condenser Pipeway Room:					
1	Iso. Cond. Line 1303-12"	558'	180°	X	X	
2	Iso. Cond. Line 1303-12"	568'	180°	X	X	
3	Iso. Cond. Line 1302-14"	580'	195°	X	X	

*Modifications to this table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment request.

H. Recirculation Pump Flow Mismatch

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

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

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

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all hydraulic snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments.

Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.6.1 as being in high radiation areas need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lbs. are exempt from the functional testing requirements because of the impracticability of testing such large units.

1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Suppressors.

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. 22
License No. 25

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

   A. The applications for amendment by the Commonwealth Edison Company (the licensee) dated December 10, 1974, October 1, 1975, and March 17, 1976, comply 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.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 22

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following existing pages of the Technical Specifications with the attached revised pages bearing the same numbers, except as otherwise indicated. Changed areas on the revised pages are shown by marginal lines.

REMOVE	INSERT
Pages iii of Table of Contents	Pages iii of Table of Contents
-	91b (new page)
-	91c (new page)
-	91d (new page)
-	91e (new page)
99a	99a
-	99b (new page)

3.6 LIMITING CONDITION FOR OPERATION

I. Shock Suppressors (Snubbers)

1. During all modes of operation except cold shutdown and refuel, all safety-related snubbers listed in Table 3.6.1 shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

3. If the requirements of 3.6.I.1 and 3.6.I.2 can not 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.

5. Snubbers may be added to safety related systems without prior license amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next license amendment request.

4.6 SURVEILLANCE REQUIREMENT

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Table 3.6.1.

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability in accordance with the following schedule:

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. The initial inspection shall be performed within six months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.I.1, it shall be assumed that the facility had been on a six-month inspection interval.

4. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lbs. need not be functionally tested.

TABLE 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Azimuth	Snubber in High* Radiation Area During Shutdown	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
2	Torus Ring Header 1501-24"	483'	83°			X
3	Torus Ring Header 1501-24"	483'	74°			X
4	Torus Ring Header 1501-24"	483'	38°			X
5	Torus Ring Header 1501-24"	483'	29°			X
7	Torus Ring Header 1501-24"	483'	331°			X
8	Torus Ring Header 1501-24"	483'	322°			X
9	Torus Ring Header 1501-24"	483'	286°			X
10	Torus Ring Header 1501-24"	483'	277°			X
12	Torus Ring Header 1501-24"	483'	218°			X
13	Torus Ring Header 1501-24"	483'	209°			X
15	Torus Ring Header 1501-24"	483'	151°			X
16	Torus Ring Header 1501-24"	483'	142°			X
1	Drywell Recirc. Motor 3B-202	524'	328°	X	X	
2	Drywell Recirc. Motor 3B-202	524'	302°	X	X	
3	Drywell Recirc. Motor 3B-202	524'	315°	X	X	
4	Drywell Recirc. Motor 3A-202	524'	148°	X	X	
5	Drywell Recirc. Motor 3A-202	524'	122°	X	X	
6	Drywell Recirc. Motor 3A-202	524'	135°	X	X	
7	Drywell Recirc. Pump 3B-202	512'	326°	X	X	
8	Drywell Recirc. Pump 3B-202	512'	304°	X	X	
9	Drywell Recirc. Pump 3B-202	507'	315°	X	X	
10	Drywell Recirc. Pump 3A-202	512'	124°	X	X	
11	Drywell Recirc. Pump 3A-202	512'	146°	X	X	
12	Drywell Recirc. Pump 3A-202	507'	135°	X	X	

Table 3.6.1 (Continued)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Azimuth	Snubber in High* Radiation Area During Shutdown	Snubbers Inaccessible During Normal Operations	Snubbers Accessible During Normal Operations
13	Drywell Recirc. Line 201B-28"	507'	305°	X	X	
14	Drywell Recirc. Line 201A-28"	507'	105°	X	X	
15	Drywell LPCI Line 1506-16"	513'	256°	X	X	
16	Drywell LPCI Line 1519-16"	513'	95°	X	X	
17	Drywell Recirc. Header 201B-22"	533'6"	195°	X	X	
18	Drywell HPCI Line 2305-10"	531'	117°	X	X	
19	Drywell HPCI Line 2305-10"	533'	112°	X	X	
20	Drywell HPCI Line 2305-10"	533'	108°	X	X	
21	Drywell Recirc. Header 201A-22"	533'6"	22°	X	X	
22	Drywell HPCI Line 2305-10"	550'	121°	X	X	
23	Drywell Cleanup Line 1201-8"	537'6"	84°	X	X	
24	Drywell Feedwater Line 3204D-12"	538'	106°	X	X	
25	Drywell Cleanup Line 1201-8"	537'6"	78°	X	X	
26	Drywell Feedwater Line 3204A-18"	537'6"	66°	X	X	
27	Drywell Cleanup Line 1201-8"	538'6"	60°	X	X	
28	Drywell Feedwater Line 3204C-12"	540'	73°	X	X	
29	Drywell Core Spray Line 1404-10"	573'	231°	X	X	
30	Drywell Core Spray Line 1403-10"	563'	336°	X	X	
31	Drywell HPCI Line 2305-10"	563'	140°	X	X	
32	Drywell Target Rock Valve 203-3A	542'6"	14°	X	X	
33	Drywell Target Rock Valve 203-3A	542'2"	31°	X	X	
34	Drywell Target Rock Valve 203-3A	540'	19°	X	X	
35	Drywell Target Rock Valve 203-3A	540'6"	34°	X	X	
	Isolation Condenser Pipeway Room:					
1	Iso. Cond. Line 1303-12"	558'	180°	X	X	
2	Iso. Cond. Line 1303-12"	568'	180°	X	X	
3	Iso. Cond. Line 1302-14'	580'	195°	X	X	

*Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

H. Recirculation Pump Flow Mismatch

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

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

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

ECCS performance during reactor operation with one recirculation loop out of service has not been analyzed. Therefore, sustained reactor operation under such conditions is not permitted.

I. Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all hydraulic snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments.

Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.6.1 as being in high radiation areas need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lbs. are exempt from the functional testing requirements because of the impracticability of testing such large units.

1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Suppressors.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 25 AND 22 TO
LICENSE NOS. DPR-19 AND DPR-25

COMMONWEALTH EDISON COMPANY

DOCKET NOS. 50-237 AND 50-249

INTRODUCTION

During the summer of 1973, inspections at two reactor facilities revealed a high incidence of inoperable hydraulic shock suppressors (snubbers) manufactured by Bergen Paterson Pipesupport Corporation. As a result of those findings, the Office of Inspection and Enforcement required each operating reactor licensee to immediately inspect all Bergen Paterson snubbers utilized on safety systems and to reinspect them 45 to 90 days after the initial inspection. Snubbers supplied by other manufacturers were to be inspected on a lower priority basis.

Since a long term solution to eliminate recurring failures was not immediately available, the Division of Reactor Licensing sent a letter dated October 1, 1973, to operating facilities (including Dresden Unit Nos. 2 and 3) utilizing Bergen Paterson snubbers specifying continuing surveillance requirements and requesting a submittal within one year of proposed Technical Specifications for a snubber surveillance program. On December 10, 1974, October 1, 1975, and March 17, 1976, Commonwealth Edison proposed Technical Specifications for hydraulic snubbers at Dresden Nuclear Power Station Unit Nos. 2 and 3. During our review of the proposed changes, we found that certain modifications were necessary. These modifications were discussed with Commonwealth Edison and have been incorporated into the proposed Technical Specifications.

EVALUATION

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal movement during startup and shutdown.

The consequence of an inoperable snubber is an increase in the probability of structural damage to piping resulting from a seismic or other postulated event which initiates dynamic loads. It is, therefore, necessary that snubbers installed to protect safety system piping be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective snubbers at reactor facilities has shown that the high incidence of failures observed in the summer of 1973 was caused by severe degradation of seal materials and subsequent leakage of the hydraulic fluid. The basic seal materials used in Bergen Paterson snubbers were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories (Reference 1) established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations. Although the molded polyurethane exhibited greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installation.

An extensive seal replacement program has been carried out at many reactor facilities. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incidence of snubber failures, some failures continue to occur. These failures have generally been attributed to faulty snubber assembly and installation, loose fittings and connections and excessive pipe vibrations. The failures have been observed in both PWRs and BWRs and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of snubber failures, we have concluded that snubber operability and surveillance requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

The proposed Technical Specifications, as modified, and Bases provide additional assurance of satisfactory snubber performance and reliability. The specifications require that snubbers be operable during reactor operation and prior to startup. Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repair or replacement of defective units before the reactor must be shut down. The licensee will be expected to commence repair

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

or replacement of a failed snubber expeditiously. However, the allowance of 72 hours is consistent with that provided for other safety-related equipment and provides for remedial action to be taken in accordance with 10 CFR 50.36(c)(2). Failure of a pipe, piping system, or major component would not necessarily result from the failure of a single snubber to operate as designed, and even a snubber devoid of hydraulic fluid would provide support for the pipe or component and reduce pipe motion. The likelihood of a seismic event or other initiating event occurring during the time allowed for repair or replacement is very small. Considering the large size and difficult access of some snubber units, repair or replacement in a shorter time period is not practical. Therefore, the 72 hour period provides a reasonable and realistic period for remedial action to be taken.

An inspection program is specified to provide additional assurance that the snubbers remain operable. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The longest inspection interval allowed in the Technical Specifications after a record of no snubber failures has been established is nominally 18 months. Experience at operating facilities has shown that the required surveillance program should provide an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Snubbers containing seal material which has not been demonstrated to be compatible with the operating environment are required to be inspected every 31 days until the compatibility is established or an appropriate seal change is completed.

To further increase the level of snubber reliability, the Technical Specifications require functional tests once each refueling cycle. The tests will verify proper piston movement, lock up and bleed.

We have concluded that the proposed additions to the Technical Specifications, as modified, increase the probability of successful snubber performance, increase reactor safety and we therefore find them accceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: September 29, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-237 AND 50-249

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 25 and 22 to Facility Operating License Nos. DPR-19 and DPR-25, respectively, issued to the Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of the Dresden Nuclear Power Station Units 2 and 3 (the facilities), located in Grundy County, Illinois. The amendments are effective as of their date of issuance.

These amendments revised the Technical Specifications to identify safety-related shock suppressors (snubbers) and include requirements regarding operability and surveillance of these snubbers.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since these amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for the amendments dated December 10, 1974, October 1, 1975, and March 17, 1976, (2) Amendment No. 25 to License No. DPR-19, (3) Amendment No. 22 to License No. DPR-25, and (4) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60451. A single copy of items (2) through (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 29th day of September, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

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