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Docket No. 50-331

Iowa Electric Light & Power Company
ATTN: Mr. Duane Arnold, President
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 to Facility License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications and is in response to your applications dated September 24, 1975, as superseded by letter dated May 17, 1976.

This amendment to the Technical Specifications adds a section which will incorporate a program for the surveillance of shock suppressors (snubbers) on safety related systems.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24
2. Safety Evaluation
3. Federal Register Notice

cc: See page 2

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

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-49

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

   A. The applications for amendment by Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensees) dated September 24, 1975, as superseded by letter dated May 17, 1976, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and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 a change 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 26, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace pages ii, vi, 3.6-10a and 3.6-32 with the attached revised pages. Add pages 3.6-10b, 3.6-32a, 3.6-32b, 3.6-41, 3.6-42, 3.6-43, 3.6-44 and 3.6-45.

Amendment No. ~~15~~, 24

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	7. At the end of each 10-year inspection interval, a report shall be submitted to the NRC that defines which of the following examination categories, if any, could not be completed: a. Class 1 components - Categories N, L-2, and M-2. b. Class 2 components - Category C-H.

H. Shock Suppressors (Snubbers)

1. During all modes of operation, except Cold Shutdown and Refuel, all safety related snubber listed in Tables 4.6-3 and 4.6-4 shall be operable, except as noted in 3.6.H.2 through 3.6.H.4 below.

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.H.1 and 3.6.H.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

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

H. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all hydraulic snubbers listed in Tables 4.6-3 and 4.6-4:

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 connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or 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%
$\geq$ 8	31 days ± 25%

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

5 **Snubbers may be added to safety related systems without prior License Amendment to Tables 4.6-3 or 4.6-4 provided that a revision to Table 4.6-3 or 4.6-4 is included with the next License Amendment request.**

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

Snubbers are categorized in two groups, "accessible and inaccessible" based on their accessibility for inspection during reactor operation. These two groups will 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 ± 25% from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.H.1, it shall be assumed that the facility has been on a 6-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 (10) hydraulic snubbers shall be so tested until notmore failures are found or all units per category tested have been tested. Snubbers of rated capacity greater than 50,000 lb. need not be functionally tested.**

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Appendix J of the DAEC FSAR provides further detail as to the inspection program planned for the DAEC.

3.6.H & 4.6.H BASES:

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 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.H.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 will be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten (10) snubbers represent an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers in high radiation areas or those especially difficult to remove need not be selected for functional tests provided operability was previously verified. Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the impracticability of testing such large units.

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

TABLE 4.6-3

HYDRAULIC SNUBBERS ACCESSIBLE DURING NORMAL OPERATION

Identification No.	System	Bldg. Location	Vendor Dwg. No.
GBC-1-SS-56	RHR Service Water	Reactor	6156
GBC-1-SS-57	RHR Service Water	Reactor	6157
GBC-2-SS-62	RHR Service Water	Reactor	6162
HCC-8-SS-11	Core Spray Pump Suction	Reactor	1787
HCC-8-SS-12	Core Spray Pump Suction	Reactor	1788/1
DCA-14-SS-73	RWCU Return to FW Line	Reactor	8518/1
EBB-16-SS-231	RHR	Reactor	2084
EBB-16-SS-232(2 ea.)	RHR	Reactor	2085
EBB-16-SS-233	RHR	Reactor	2086
EBB-16-SS-234(2 ea.)	RHR	Reactor	2087
GBB-3-SS-235	RHR	Reactor	2088
GBB-3-SS-236	RHR	Reactor	2089
GBB-3-SS-237	RHR	Reactor	2090
GBB-3-SS-238	RHR	Reactor	2091
GLE-8-SS-239	RHR	Reactor	2092
GLE-8-SS-240	RHR	Reactor	2093
GBB-10-SS-241	RHR	Reactor	2094
GBB-10-SS-242(2 ea.)	RHR	Reactor	2095
GBB-10-SS-243	RHR	Reactor	2096
GBB-4-SS-210	RHR	Reactor	2063
GBB-4-SS-211	RHR	Reactor	2064
GBB-4-SS-212	RHR	Reactor	2065
GBB-4-SS-213	RHR	Reactor	2066
GBB-16-SS-214	RHR	Reactor	2067
GBB-5-SS-215	RHR	Reactor	2068
GBB-4-SS-216 (2 ea.)	RHR	Reactor	2069
GBB-4-SS-217 (2 ea.)	RHR	Reactor	2070
HBB-21-SS-218(2 ea.)	RHR	Reactor	2071
HBB-23-SS-219	RHR	Reactor	2072
HBB-23-SS-220	RHR	Reactor	2073
HBB-24-SS-221	RHR	Reactor	2074
HBB-24-SS-222	RHR	Reactor	2075
GBB-7-SS-223	RHR	Reactor	2076
GBB-7-SS-224	RHR	Reactor	2077
GBB-6-SS-225	RHR	Reactor	2078
GBB-6-SS-226	RHR	Reactor	2079
HBB-24-SS-227(2 ea.)	RHR	Reactor	2080
HBB-24-SS-228(2 ea.)	RHR	Reactor	2081
HBB-24-SS-229	RHR	Reactor	2082
HBB-29-SS-199	RHR	Reactor	2052
HBB-30-SS-205	RHR	Reactor	2058
HBB-30-SS-206	RHR	Reactor	2059
HBB-30-SS-245	RHR	Reactor	2098

(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 4.6-3 (cont.)

Identification No.	System	Bldg. Location	Vendor Dwg. No.
HBB-7-SS-17	RCIC Turbine Exhaust	Reactor	1677
HBB-7-SS-18	RCIC Turbine Exhaust	Reactor	1678
HBB-7-SS-19 (2 ea.)	RCIC Turbine Exhaust	Reactor	1680
GBB-13-SS-16	Core Spray	Reactor	1792
GBB-14-SS-20	Core Spray	Reactor	1796
HBB-2-SS-7	Core Spray	Reactor	1783
HBB-2-SS-8	Core Spray	Reactor	1784
HBB-1-SS-9	Core Spray	Reactor	1785
HBB-1-SS-10	Core Spray	Reactor	1786
EBB-14-SS-13	HPCI Steam Supply	Reactor	1579
EBB-14-SS-14	HPCI Steam Supply	Reactor	1580
EBB-14-SS-15	HPCI Steam Supply	Reactor	1581
EBB-14-SS-16	HPCI Steam Supply	Reactor	1582
EBB-14-SS-16A	HPCI Steam Supply	Reactor	1582A
HBB-6-SS-20	HPCI Turbine Exhaust	Reactor	1586
HBB-6-SS-22	HPCI Turbine Exhaust	Reactor	1588
HBD-31-SS-71	Emergency Service Water	Reactor	6171
HBD-31-SS-101	Emergency Service Water	Reactor	6201
HBB-25-SS-178	Fuel Pool to RHR	Reactor	2031
GBD-29-SS-12	Aux. Boiler Stm. to HPCI	Reactor	1578/1
DCB-2-SS-78	RWCU	Reactor	8523

(2 ea.) - Indicates there are 2 snubbers with that number.

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

TABLE 4.6-4

SNUBBERS INACCESSIBLE DURING NORMAL OPERATION

Identification No.	System	Location	Vendor Dwg. No.
DLA-5-SS-10	RHR	Drywell	6010A/1
DLA-5-SS-11	RHR	Drywell	6011
DLA-6-SS-12	RHR	Drywell	6012
DLA-6-SS-13	RHR	Drywell	6013
DLA-4-SS-14	RHR	Drywell	6014
DLA-4-SS-15	RHR	Drywell	6015
DBA-4-SS-35	RCIC	Drywell	6035
DBA-4-SS-36	RCIC	Drywell	6036
DLA-3-SS-1	HPCI Steam Supply	Drywell	6001
DLA-3-SS-2	HPCI Steam Supply	Drywell	6002
DLA-3-SS-3	HPCI Steam Supply	Drywell	6003
DBA-6-SS-29	CRD	Drywell	6029
DBA-6-SS-30	CRD	Drywell	6030
DCA-6-SS-48	RWCU	Drywell	6048
DCA-6-SS-49	RWCU	Drywell	6049
DCA-6-SS-50	RWCU	Drywell	6050
DBA-4-SS-34	RCIC Steam Supply	Drywell	6034
DBA-5-SS-31	Head Spray	Drywell	6031
DBA-5-SS-37	Head Spray	Drywell	6037
DBA-5-SS-38	Head Spray	Drywell	6038
DBA-5-SS-47	Head Spray	Drywell	6047
DLA-2-SS-4	RHR	Drywell	6004
DLA-2-SS-5	RHR	Drywell	6005
DLA-2-SS-6	RHR	Drywell	6006
DLA-2-SS-7	RHR	Drywell	6007
DLA-2-SS-8	RHR	Drywell	6008
DLA-2-SS-9	RHR	Drywell	6009
DBA-7-SS-71	RCIC to FW Line	Reactor	8516
DCA-14-SS-72	RWCU to FW Line	Reactor	8517
GBC-6-SS-16	Main Stm. Relief Valve Discharge	Drywell	6016
GBC-6-SS-17	" " "	Drywell	6017
GBC-7-SS-18	" " "	Drywell	6018
GBC-7-SS-19	" " "	Drywell	6019
GBC-8-SS-20	" " "	Drywell	6020
GBC-8-SS-21	" " "	Drywell	6021
GBC-8-SS-44	" " "	Drywell	6044
GBC-8-SS-45	" " "	Drywell	6045
GBC-8-SS-46	" " "	Drywell	6046
GBC-9-SS-22	" " "	Drywell	6022
GBC-9-SS-23	" " "	Drywell	6023
GBC-9-SS-41	" " "	Drywell	6041
GBC-9-SS-42 (2 ea.)	" " "	Drywell	6042

(2 ea.) - Indicates there are 2 snubbers with that number.

TABLE 4.6-4 (cont.)

Identification No.	System	Location	Vendor Dwg. No.
GBC-9-SS-43	Main Stm. Relief Valve Discharge	Drywell	6043
GBC-10-SS-24	" " "	Drywell	6024
GBC-10-SS-25	" " "	Drywell	6025
GBC-10-SS-39	" " "	Drywell	6039
GBC-10-SS-40 (2 ea.)	" " "	Drywell	6040
GBC-11-SS-26	" " "	Drywell	6026
GBC-11-SS-27	" " "	Drywell	6027
GBC-11-SS-32	" " "	Drywell	6032
GBC-11-SS-33	" " "	Drywell	6033
SSB-1-MS	Main Steam	Drywell	GE-BP 405 Rev 2
SSB-2-MS	Main Steam	Drywell	GE-BP 406 Rev 3
SSC-1-MS	Main Steam	Drywell	GE-BP 407 Rev 2
SSC-2-MS	Main Steam	Drywell	GE-BP 408 Rev 2
SSA-1-MS	Main Steam	Drywell	GE-BP 401 Rev 1
SSA-2-MS	Main Steam	Drywell	GE-BP 402 Rev 1
SSD-1-MS	Main Steam	Drywell	GE-BP 403 Rev 1
SSD-2-MS	Main Steam	Drywell	GE-BP 404 Rev 1
SSA-1	Recirc	Drywell	GE-BP 201 Rev 2
SSB-1	Recirc	Drywell	GE-BP 202 Rev 2
SSA-2	Recirc	Drywell	GE-BP 203 Rev 1
SSB-2	Recirc	Drywell	GE-BP 204 Rev 1
SSA-3	Recirc	Drywell	GE-BP 205 Rev 1
SSB-3	Recirc	Drywell	GE-BP 206 Rev 1
SSA-4	Recirc	Drywell	GE-BP 207 Rev 1
SSB-4	Recirc	Drywell	GE-BP 208 Rev 1
SSA-5	Recirc	Drywell	GE-BP 209 Rev 1
SSB-5	Recirc	Drywell	GE-BP 210 Rev 1
SSA-6	Recirc	Drywell	GE-BP 211 Rev 1
SSB-6	Recirc	Drywell	GE-BP 212 Rev 1
SSA-7	Recirc	Drywell	GE-BP 213 Rev 1
SSB-7	Recirc	Drywell	GE-BP 214 Rev 1
SSA-8	Recirc	Drywell	GE-BP 215 Rev 1
SSB-8	Recirc	Drywell	GE-BP 216 Rev 1
SSA-9	Recirc	Drywell	GE-BP 217 Rev 1
SSB-9	Recirc	Drywell	GE-BP 218 Rev 1
SSA-10	Recirc	Drywell	GE-BP 219 Rev 1
SSB-10	Recirc	Drywell	GE-BP 220 Rev 1
SSA-11	Recirc	Drywell	GE-BP 221 Rev 1
SSB-11	Recirc	Drywell	GE-BP 222 Rev 1

(2 ea.) - Indicates there are 2 snubbers with that number.

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

TABLE 4.6-5

SNUBBERS IN HIGH RADIATION AREA DURING SHUTDOWN
AND/OR ESPECIALLY DIFFICULT TO REMOVE

None

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO LICENSE NO. DPR-49

IOWA ELECTRIC LIGHT & POWER COMPANY

DUANE ARNOLD ENERGY CENTER NO. 1

DOCKET NO. 50-331

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.

By letters dated July 8, 1975 and December 24, 1975, we provided model Technical Specifications for snubber surveillance and requested Iowa Electric Light and Power Company (IELP) to submit an application for license amendment. Iowa Electric submitted proposed Technical Specifications by letter dated September 24, 1975, as superseded by letter dated May 17, 1976. During our review we found that certain modifications were necessary. These modifications were discussed with the licensee on November 1, 1976 and have been included in the proposed Technical Specifications.

The proposed change to the Technical Specification adds new Sections 3.6.H and 4.6.H which provide the Limiting Conditions for Operation and Surveillance Requirements associated with safety related shock suppressors.

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 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 PWR's and BWR's 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 attached Technical Specifications 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 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

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

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 Technical Specifications, as modified, increase the probability of successful snubber performance, increase reactor safety and we therefore find them acceptable.

Environmental Consideration

We have determined that this amendment does 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 amendment involves 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 appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 26, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, which revised Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County, Iowa. The amendment is effective as of its date of issuance.

The amendment to the Technical Specifications adds a section which will incorporate a program for the surveillance of shock suppressors (snubbers) on safety related systems.

The applications for the amendment comply 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment dated September 24, 1975, as superseded by letter dated May 17, 1976, (2) Amendment No. 24 to License No. DPR-49, and (3) the Commission's 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 Cedar Rapids Public Library, 426 Third Avenue, S. E., Cedar Rapids, Iowa 52401

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 26th day of November, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors