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Docket No. 50-331

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

Gentlemen:

In response to your requests dated December 22, 1975, November 20, 1975 and February 24, 1976, the Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center.

The amendment consists of changes in the Technical Specifications that will (1) add limiting conditions for operation and surveillance requirements for a new main steam line isolation valve leakage control system (2) update the inservice inspection requirements for Class 1 and Class 2 components, and (3) remove the lower limit for closure times of the purge supply, drywell purge outlet, and torus purge outlet power operated isolation valves.

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

Sincerely,

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

Enclosures:
1. Amendment No. 18 to License DPR-49
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

OFFICE➤	ORB#3	ORB#3 WM	~~ORB#3~~ OELD	ORB#3		
SURNAME➤	CParrish CP	WPaulson:acr		GLear G		
DATE➤	3/ 9 /76	3/ 10 /76	3/ 11 /76	3/ 10 /76		

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

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Office for Planning and Programming
523 East 12th Street
Des Moines, Iowa 50319

Mr. Dudley Henderson
Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

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. 18
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensees) dated December 22, 1975, November 20, 1975 and February 24, 1976, 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

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: March 17, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace pages 3.6.8, 3.6-9, 3.6-10, 3.6-30, 3.6-31, 3.6-32, 3.6-36 through 3.6-40, 3.7-16, 3.7-25, 3.7-43 and 3.7-44 with the attached revised pages. No change has been made on pages 3.6-35, 3.7-19, 3.7-20 and 3.7-26. Add pages 3.6-10a, 3.7-19a and 3.7-49a.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3. Following 1-pump operation, the discharge valve of the lower speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

G. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standard throughout the life of the plant.

G. Structural Integrity

1.a. Nuclear Class I Components - Components within the reactor coolant pressure boundary (as defined in Article IS-120 of the ASME Boiler and Pressure Vessel Code) will be pressure tested prior to startup following each reactor refueling outage. During the pressure test, components will be inspected for leakage without removal of insulation. The test pressure and temperature will be maintained for at least four hours prior to the final leakage inspection. The test pressure will not be less than the system nominal operating pressure at 100% rated reactor power. The pressure test will be conducted at a vessel temperature above the nil ductility temperature of the vessel.

Near the end of each inspection interval, one system pressure test will be upgraded to a system hydrostatic pressure test. The hydrostatic test will be identical with the pressure test, except that the minimum test pressure will be higher and the test will be witnessed by an authorized inspector. The test pressure will not be less than 1.08 times the system nominal operating pressure as required by Subsection IS-522 of the Winter 1972 Addenda to Section XI of the ASME Boiler and Pressure Vessel Code.

b. Nuclear Class II Components - Near the end of each inspection interval, the following systems (as defined in Subsection ISC-261 of the Winter 1972 Addendum to Section XI of the ASME Boiler and Pressure

Vessel Code) will be subjected to an operational pressure test.

1) RHR System
2) Core Spray System
3) HPCI System
4) RCIC System
5) Standby Liquid Control System

Components will be subjected to normal operating pressure by operation of system pumps or by remote pressurization.

2. The visual examinations above shall be conducted under the guidelines of the procedures of Articles IS-211 (Nuclear Class I) and ISC-210 (Nuclear Class II) of Section XI of the ASME Boiler and Pressure Vessel Code.

This examination (which need not require removal of insulation) shall be performed by inspecting (a) the exposed surface and joints of insulations, and (b) the floor areas (or equipment) directly underneath these components.

At locations where reactor coolant leakage is normally expected and collected (e.g., valve stems, etc.), the examination shall verify that the leakage collection system is operative and that there is no evidence of leakage in the collection system.

3. The source of any unacceptable reactor coolant leakage detected by the above examination shall be located by the removal of insulation where necessary and the following corrective measures applied:

Normally expected leakage from component parts (e.g., valve stems) shall be minimized by

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	appropriate repairs and maintenance procedures.
	Leakage from through-wall flaws in the pressure-retaining membrane of a component shall be eliminated, either by corrective repairs or by component replacement. Such repairs shall conform with the requirements of Article IS-400 of Section XI of the ASME Boiler and Pressure Vessel Code.
	4. The nondestructive inspections listed in Tables 4.6-1 and 4.6-2 shall be performed as specified.
	5. The structural integrity of the primary reactor coolant pressure system boundary and the reactor coolant associated auxiliary systems shall be maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence as a result of the inspection listed in Tables 4.6-1 and 4.6-2 that unacceptable defects have initiated or indications have grown significantly shall be investigated, including evaluation of comparable areas of the primary system. In the event further unacceptable structural defects are revealed, all remaining system components or piping in this category shall be examined to the extent practical as specified in that examination category.
	6. Detailed records of each inspection, including the preoperational base line inspection, shall be maintained to allow comparison and evaluation or future inspection. The records shall conform to the requirements of IS-600 of Section XI of the ASME Boiler and Pressure Vessel Code.

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

3.6.G & 4.6.G BASES:

REACTOR COOLANT SYSTEM

Structural Integrity

A pre-service inspection of Nuclear Class I Components was conducted to assure freedom from defects greater than code allowance; in addition, this served as a reference base for future inspections. Prior to operation, the reactor coolant system as described in Article IS-120 of Section XI of the ASME Boiler and Pressure Vessel Code was inspected to provide assurance that the system was free of gross defects. In addition, the facility was designed such that gross defects should not occur throughout plant life. The pre-service inspection program was based on the 1970 Section XI of the ASME Code for in-service inspection. This inspection plan was designed to reveal problem areas (should they occur) before a leak in the coolant system could develop. The program was established to provide reasonable assurance that no LOCA would occur at the DAEC as a result of leakage or breach of pressure-containing components and piping of the reactor coolant system, portions of the ECCS, and portions of the reactor coolant associated auxiliary systems.

A pre-service inspection was not performed on Nuclear Class II Components because it was not required at that stage of DAEC construction when it would have been used. For these components, shop and in-plant examination records of components and welds will be used as a basis for comparison with in-service inspection data.

The engineering and design effort associated with the Duane Arnold Energy Center predates the availability of the ASME Inspection Code. However, this Code, including subsequent Addendum through the Winter 1972 Addenda, dated December 31, 1972, has been used as a guide in the preparation of the DAEC In-service Inspection Plan for Nuclear Class I and Class II Components, and maximum access has been provided to the extent drywell design and radiation levels permit.

The examination program for Nuclear Class I Components includes those portions of the pressure containing components up to and including the outermost containment isolation valve which could isolate the primary systems in the event of a loss-of-coolant accident (LOCA). The examination program for Nuclear Class II Components includes portions of Nuclear Class II Systems as required by Subsection ISC-261 of the Winter 1972 Addenda to the In-service Inspection Code. The Examination Program assumes that examinations can be performed without the necessity of unloading the reactor core solely for the purpose of conducting examinations.

Visual inspections for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC and is delineated by Section XI of the ASME Code. These studies show that it requires thousands of stress cycles at stresses beyond those expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results, only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

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.

TABLE 4.6-1 (Continued)

NUCLEAR CLASS I ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING 3-1/3 YEAR INTERVAL %	TENTATIVE INSPECTION DURING 10 YEAR INTERVAL %	REMARKS
1.4	D	Nozzle-to-Vessel Welds and Nozzle Inside Radiused Section	Volumetric	25 to 33-1/3%	100%	It is planned that the nozzles will be inspected from the vessel O.D. Inspection of the 2" instrument nozzle immediately above the core will be made provided: 1. Radiation levels permit the inspections to be made without undue exposure of personnel. or 2. Techniques or equipment are developed that allow inspection to be made without undue personnel exposure.
1.5	E-1	Vessel Penetrations	Volumetric	0	0	All penetrations are Category E-2
1.6	E-2	Vessel Penetrations Including CRD's, In-Core Monitors	Visual	8%	25%	None
1.7	F	Dissimilar Metal Welds	Visual Surface and Volumetric	25 to 33-1/3%	100%	Dissimilar metal welds will be examined at the same time and sequence as the nozzle-to-vessel welds. See Item 1.4 Category D remarks concerning 2" n[illegible]s.
1.8	G-1	Closure Studs and Nuts	Volumetric and Visual or Surface	25 to 33-1/3%	100%	100% visual each year for thread damage.
1.9	G-1	Ligaments Between Threaded Stud Holes	Volumetric	25 to 33-1/3%	100%	The ligaments will be examined at the same time as the flange weld of Item 1.3.
1.10	G-1	Closure Washers and Bushings	Visual	33%	100%	None

TABLE 4.6-1 (Continued)

NUCLEAR CLASS I ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING 3-1/3 YEAR INTERVAL %	TENTATIVE INSPECTION DURING 10 YEAR INTERVAL %	REMARKS
1.11	G-2	Pressure Retaining Bolting	Visual	0	0	All bolting is over 2" in diameter and is in category G-1.
1.12	H	Integrally Welded Vessel Supports	Volumetric	0	10%	None
1.13	I-1	Closure Head Cladding	Surface and Visual	0	0	The vessel head is not clad.
1.14	I-1	Vessel Cladding	Visual	2 Patches	6 Patches	None
1.15	N	Interior Surfaces and Internals and Integrally Welded Internal Supports	Visual	See Remarks	See Remarks	Items made accessible by the removal of components during normal refueling will be examined at the first refueling outage and during subsequent outages at approximately 3 year intervals.
II. PIPING PRESSURE BOUNDARY						
4.1	F	Dissimilar Metal Welds in Branch Piping	Visual and Surface and Volumetric	25 to 33-1/3%	100%	The dissimilar metal welds in the piping systems will be identified for the pre-operational inspection.
4.2	G-1	Pressure Retaining Bolting 2" and larger	Visual and Volumetric	0	0	All bolting is less than 2 inches in diameter
4.3	G-2	Pressure Retaining Bolting Less Than 2"	Visual	25 to 33-1/3%	100%	None
4.4	J-1	Circumferential and Longitudinal Pipe Welds.	Visual and Volumetric.	6-1/4 to 8-1/3%	25%	The examinations include the circumferential and one foot of the longitudinal welds on each side of the butt welds. Due to inaccessibility the drain line welds will be excluded.

TABLE 4.6-1 (Continued)

NUCLEAR CLASS I ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING 3-1/3 YEAR INTERVAL %	TENTATIVE INSPECTION DURING 10 YEAR INTERVAL %	REMARKS
4.5	J-1	Branch Pipe Connection Welds Exceeding 4" Nom. Pipe Size	Visual and Volumetric	8%	25%	None
4.6	J-1	Socket Welds	Visual and Surface	8%	25%	None
4.7	J-1	Branch Pipe Connection Welds, 4" Nom. Pipe Size and Smaller	Visual and Surface	8%	25%	None
4.8	J-2	Circumferential and Longitudinal Pipe Welds and Branch Pipe Connection Welds	Visual	25 to 33-1/3%	100%	None
4.9	K-1	Integrally Welded Supports	Visual and Volumetric	8%	25%	None
4.10	K-2	Piping Supports and Hangers	Visual	25 to 33-1/3%	100%	None
III.	PUMP PRESSURE BOUNDARY					
5.1	L-1	Pump Casing Welds	Visual	0	0	The pumps do not have pressure retaining casing welds.
5.2	L-2	Pump Casings	Visual	0	See Remarks	At least one of every type of pump of similar design and service will be inspected. Internal surfaces will be examined only when a pump is disassembled for other reasons.
5.3	F	Dissimilar Metal Welds	Visual Surface and Volumetric	0	0	The pumps do not have dissimilar metal welds.
5.4	G-1	Pressure Retaining Bolting	Visual and Volumetric	33%	100%	None

TABLE 4.6-1 (Continued)

NUCLEAR CLASS I ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING 3-1/3 YEAR INTERVAL %	TENTATIVE INSPECTION DURING 10 YEAR INTERVAL %	REMARKS
5.5	G-2	Pressure Retaining Bolting	Visual	0	0	All bolting is over 2" in diameter and is in Category G-1.
5.6	K-1	Integrally Welded Supports	Visual and Volumetric	6-1/4 to 8-1/3%	25%	None
5.7	K-2	Supports and Hangers	Visual	33%	100%	None
IV. VALVE PRESSURE BOUNDARY						
6.1	M-1	Valve Body Welds	Visual and Volumetric	0	0	There are no pressure retaining welds in valve bodies.
6.2	M-2	Valve Bodies	Visual	0	See Remarks	At least one of every type of valve of similar design and service will be inspected Internal surfaces will be examined only when a valve is disassembled for other reasons. In addition, the valves in the main recircu- lation piping loops cannot be inspected with out removing the fuel from the core and drai ing the reactor system.
6.3	F	Valve-to-Safe End Welds	Surface Visual and Volumetric	0	0	There are no dissimilar metal weld the valves.
6.4	G-1	Pressure Retaining Bolting	Visual and Volumetric	0	0	The bolting is less than 2".
6.5	G-2	Pressure Retaining Bolting	Visual	25 to 33-1/3%	100%	None
6.6	K-1	Integrally Welded Supports	Visual and Volumetric	0	0	No integrally welded supports on valves.
6.7	K-2	Supports and Hangers	Visual	25%	100%	None

TABLE 4.6-2

NUCLEAR CLASS II ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING FIRST 10 YEAR INTERVAL	TENTATIVE INSPECTION DURING SERVICE LIFETIME	REMARKS
I. CLASS II PRESSURE VESSELS						
C1.1	C-A	Circumferential Butt Welds	Volumetric	33%	100%	The examinations shall include at least 20 percent of each circumferential weld, uniformly distributed among three areas around the vessel circumference.
C1.2	C-B	Nozzle-to-Vessel Welds	Volumetric	50%	100%	None
C1.3	C-C	Integrally Welded Supports	Surface	0	100%	Only one inspection is required. It is scheduled for the third interval.
C1.4	C-D	Pressure Retaining Bolting	Visual and Either Surface or Volumetric	0	0	There is no vessel bolting.
II. CLASS II PIPING						
C2.1	C-F, C-G	Circumferential Butt Welds	Volumetric	25%	100%	None
C2.2	C-F, C-G	Longitudinal Weld Joints In Fittings	Volumetric	0	0	There are no longitudinal weld joints in fittings.
C2.3	C-F, C-G	Branch Pipe-to-Pipe Weld Joints	Volumetric	0	0	There are no branch pipe-to-pipe joints greater than 4" diameter.
C2.4	C-C	Integrally Welded Support-to-Pipe Welds	Surface	27%	100%	None
C2.5	C-D	Pressure Retaining Bolting	Visual and Either Surface or Volumetric	100%	100%	Only one inspection is required for this category.
C2.6	C-E	Supports and Hangers	Visual	27%	100%	None.

TABLE 4.6-2

NUCLEAR CLASS II ACCESS PROVISIONS AND EXAMINATION SCHEDULE

ITEM NO.	CATEGORY	EXAMINATION AREA	EXAMINATION METHOD	INSPECTION DURING FIRST 10 YEAR INTERVAL	TENTATIVE INSPECTION DURING SERVICE LIFETIME	REMARKS
III. CLASS II PUMPS						
C3.1	C-F, C-G	Pump Casing Welds	Volumetric	0	0	There are no pump casing welds.
C3.2	C-H	Pump Casing	Visual	See Remarks	See Remarks	At least one of every type of pump of similar design and service will be inspected. Internal surfaces will be examined only when the pumps are disassembled for other purposes.
C3.3	C-D	Pressure Retaining Bolting	Visual and Either Surface or Volumetric	50%	100%	None
C3.4	C-E	Supports and Hangers	Visual	0	0	There are no pump hangers or supports.
IV. CLASS II VALVES						
C4.1	C-F, C-G	Valve Body Welds	Volumetric	0	0	There are no valve body welds.
C4.2	C-H	Valve Bodies	Visual	See Remarks	See Remarks	At least one of every type of valve of similar design and service will be inspected. Internal surfaces will be examined only when disassembled for other purposes.
C4.3	C-D	Pressure Retaining Bolting	Visual and Either Surface or Volumetric	33%	100%	None
C4.5	C-E	Supports and Hangers	Visual	0	0	There are no valve hangers or supports.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	by partial closure and subsequent reopening.
	d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.
2. In the event any isolation valve specified in Table 3.7-3 becomes inoperable, (except for those exempted as noted in Table 3.7-3) reactor power operation may continue provided at least one valve in each line having an in operable valve shall be in the mode corresponding to the isolated condition.	2. Wherever an isolation valve listed in Table 3.7-3 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.7	4.7
E. Main Steam Isolation Valve Leakage Control System (MSIV-LCS)	E. Main Steam Isolation Valve Leakage Control System
1. The MSIV-LCS shall be operable whenever the reactor is critical or when the reactor temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.E.2 below.	1. MSIV-LCS Testing Item / Frequency a. Simulated Actuation Test — Once/Operating Cycle b. Blower Operability — Once/Month c. Motor-operated Valve Operability — Once/Month d. Heater Operability — Once/Month e. Blower Capacity — Once/Operating Cycle
2. From and after the date that one MSIV-LCS subsystem or one blower is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all active components of the other MSIV-LCS subsystems are operable.	2. When it is determined that one MSIV-LCS subsystem or one blower is inoperable, the other MSIV-LCS subsystem or blower shall be demonstrated to be operable immediately. The operable MSIV-LCS subsystems shall be demonstrated to be operable weekly thereafter.
3. If the requirements of 3.7.E cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.	

TABLE 3.7-1

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TEST REQUIREMENTS

Penetration #	Type	Description
1	Testable Gaskets	Personnel Lock Equipment Door
1	Personnel Lock	Personnel Lock Doors and Penetrations
2	Testable Gaskets	Equipment Access
4	Testable Gaskets	Head Access
6	Testable Gaskets	CRD Removal Hatch
35A-D	Testable Gaskets	TIP Drives
53	Testable Gaskets	Spare
----	Testable Gaskets	Drywell Head Flange
58 A-H	Testable Gaskets	Stabilizer Access Ports (8)
200A-B	Testable Gaskets	Torus Access Hatch
100B,C,E,F,G	Electrical Canister	(B,C,E,F) Neutron Monitoring, (G) RPV Vibration Monitoring
101A,C	Electrical Canister	(C) (A) Recirc Pump Power
103	Electrical Canister	Thermocouples
104A-D	Electrical Canister	CRD Rod Position Indicator
105B,D	Electrical Canister	(B,D) Power & Control
106A,C	Electrical Canister	(A,C) Power & Control
230B	Electrical Canister	Vacuum Breakers Electrical Cables

Amendment No. 18

TABLE 3.7-3

PRIMARY CONTAINMENT POWER OPERATED ISOLATION VALVES

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves	Maximum Operating Time (Seconds)	Normal Position	Action on Initiating Signal
1	*Main Steam Line	8	3<T<5	O	GC
1	Main Steam Line Drain	2	15	C	SC
1	Recirculation Loop Sample	2	NA	C	SC
1	Recirculation Pump Seal Purge	2	5	O	GC
3	O_2 Analyzer	20	NA	O	GC
2	Drywell Floor Drain Discharge	2	4	O	GC
2	Drywell Equipment Drain Discharge	2	4	O	GC
3	Purge Supply	3	5	C	SC
3	Drywell Purge Outlet	3	5	C	SC
3	Torus Purge Outlet	3	5	C	SC
3	Drywell and Torus Nitrogen Makeup	3	NA	O	GC
4	RHR Shutdown Cooling Supply	2	22	C	SC
3	*Containment Compressor Suction	2	25	O	GC

TABLE 3.7-3 (Continued)

PRIMARY CONTAINMENT POWER OPERATED ISOLATION VALVES

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves	Maximum Operating Time (Seconds)	Normal Position	Action on Initiating Signal
5	RWCU Supply	2	20	O	GC
5	RWCU Return	1	10	O	GC
6	Steam to HPCI Turbine	2	13	O	GC
6	HPCI Discharge to Feedwater	1	20	C	GC
6	Steam to RCIC Turbine	2	20	O	GC
6	RCIC Discharge to Feedwater	1	15	C	GC
8	Condensate from HPCI	2	NA	O	GC
8**	Condensate from RCIC	2	NA	O	GC
3	*Containment Compressor Discharge	3	NA	O	GC
7	*Reactor Building Closed Cooling Water Supply/Return	2	20	O	GC
7	*Well Cooling Water Supply/ Return	8	NA	O	GC
9	HPCI/RCIC Exhaust Vacuum Breaker	2	10	O	GC

** Low-Low Water Level Only

The MSIV-LCS system is provided to minimize the fission products which could bypass the standby gas treatment system after a LOCA. It is designed to be manually initiated after it has been determined that a LOCA has occurred and that the pressure between the MSIV's has decayed to less than 35 psig. The System is also inhibited from operating unless the inboard MSIV associated with the MSIV-LCS subsystem is closed and the reactor vessel pressure has decayed to less than 35 psig.

Checking the operability of the various components of the MSIV-LCS system monthly assures that the MSIV-LCS system will be available in the remote possibility of a LOCA. An annual capacity test of the blowers and an annual initiation of the entire system assure that the MSIV-LCS system meets its design criteria. Allowance of thirty days to return a MSIV-LCS subsystem or blower to an operable status allows operational flexibility while maintaining protective capabilities.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
2.a The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99.9% DOP removal and $\geq$ 99.9% halogenated hydrocarbon removal.	2.a The tests and sample analysis of Specification 3.7.B.2 shall be performed initially and at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show $<$ 1.0% penetration of radioactive methyl iodide at 70% R. H., 150°F, 40 ± 4 FPM face velocity with an inlet concentration of 0.5 to 1.5 mg/m^3 inlet concentration methyl iodide.	b. Cold DOP testing shall be performed after each complete or partical replacement of the HEPA filter bank or after any structural maintenance on the system housing.
	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.
c. Fans shall be shown to be capable of operation from 1800 to 4000 cfm.	d. Each circuit shall be operate with the heaters on at least 10 hours every month.
3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, continued reactor operation or fuel handling is permissible only during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components of the other standby gas treatment train shall be operable.	3. When one train of the standby gas treatment system becomes inoperable, the operable train shall be demonstrated to be operable immediately and daily thereafter.
4. If Specifications 3.7.B.1, 3.7.B.2 and 3.7.B.3 are not met, the reactor shall be placed in the cold shutdown condition and fuel handling operations shall be prohibited.	

needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the standby gas treatment system is not required.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of atleast 99.9 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99.9 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed, as the FSAR analysis shows compliance with 10 CFR 100 guidelines with an assumed efficiency of 95% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 11 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. (The design of the SGTS system allows the removal of charcoal samples from the bed directly through the use of a grain thief.) Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according

3.7.E & 4.7.E BASES:

The MSIV-LCS system is provided to minimize the fission products which could bypass the standby gas treatment system after a LOCA. It is designed to be manually initiated after it has been determined that a LOCA has occurred and that the pressure between the MSIV's has decayed to less than 35 psig. The System is also inhibited from operating unless the inboard MSIV associated with the MSIV-LCS subsystem is closed and the reactor vessel pressure has decayed to less than 35 psig.

Checking the operability of the various components of the MSIV-LCS system monthly assures that the MSIV-LCS system will be available in the remote possibility of a LOCA. An annual capacity test of the blowers and an annual initiation of the entire system assure that the MSIV-LCS system meets its design criteria. Allowance of thirty days to return a MSIV-LCS subsystem or blower to an operable status allows operational flexibility while maintaining protective capabilities.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

Introduction

By letter dated December 22, 1975, Iowa Electric Light and Power Company, Central Iowa Power Company, and Corn Belt Power Cooperative (the licensees) requested an amendment to Facility Operating License No. DPR-49. The amendment would modify the Technical Specifications for the Duane Arnold Energy Center by adding limiting conditions for operation and surveillance requirements for a new main steam line isolation valve leakage control system that is being added to the facility during the current refueling outage. The installation of the MSIV-LCS fulfills the licensees' commitment to install a system to prevent direct leakage from the containment to the environment through the main steam line isolation valves. We addressed the requirement for the installation of this system in our Safety Evaluation Report for the DAEC. In supplements Nos. 1 and 2 to the SER, we discussed the preliminary results of our review of the design concept proposed by the licensees.

The final design of the MSIV-LCS is described in Amendments 15 and 17 to the DAEC Final Safety Analysis Report and supplemented by letters dated October 3, 1975 and November 10, 1975.

By letter dated November 20, 1975, the licensees also requested an amendment to Facility Operating License No. DPR-49 that would modify the DAEC Technical Specifications to update the inservice inspection requirements for Class 1 and Class 2 components. This request for license amendment fulfills a commitment made by the licensees by letter dated January 11, 1974, in which they stated they would submit a program for updating their inservice inspection program to the provisions of Section II of the ASME Code through the Winter Addenda of 1972, to the extent practical within one year after receiving their operating license and that they would implement the approved program two years after issuance of the DAEC operating license. The licensees' proposed program is discussed in Amendment No. 16 to the DAEC Final Safety Analysis Report.

By letter dated February 24, 1976, the licensees have also requested a license amendment that would modify the DAEC Technical Specifications by removing the lower limit for closure times of the purge supply, drywell purge outlet, and torus purge outlet power operated isolation valves.

Evaluation

1. Main Steam Line Isolation Valve Leakage Control System

We reviewed the MSIV-LCS proposed by the licensees. The proposed main steam line isolation valve leakage control system (MSIV-LCS) has separate connections to each main steam line. The main steam line isolation valve (MSIV) leakage is removed from the main steam line thru a connection between the MSIV's. This system, therefore, treats only leakage thru the inboard MSIV's. The inboard MSIV leakage is directed to a manifold and then thru a blower to the standby gas treatment system.

The system is designed to be actuated manually by operators in the control room after ten minutes following a loss-of-coolant accident (LOCA). Pressure permissive interlocks are provided to prevent MSIV-LCS operation if steam line pressure is above the set point. The licensee has agreed to provide an interlock to prevent operation of that part of the system connected to an inboard MSIV which fails to close. However, if one inboard MSIV should fail to close, the remaining portion of the MSIV-LCS which processes leakage thru the other three inboard MSIV's would remain operable.

During the course of our review, we requested and the licensees agreed to upgrade the Technical Specification acceptance level for the standby gas treatment system filter system to 99.9% efficiency for the DOP and Freon tests in order to achieve a filter efficiency of 95%.

Based on our review of the licensees' proposed system, as modified, we find that with both MSIV's closed in each main steam line, and with the LCS in operation, the total accident dose would not be in excess of that previously calculated and discussed in the DAEC Safety Evaluation Report.

In addition, we performed a calculation to estimate the doses resulting from a postulated LOCA assuming that one inboard MSIV failed to close. The estimated doses at both the Exclusion Area Boundary (EAB) and at the Low Population Zone (LPZ) are shown on Table 1 and they indicate that even if one MSIV fails to close, the doses resulting from a postulated LOCA will be within the 10 CFR Part 100 guidelines. The data used to estimate the doses are shown on Table 2.

Based on our review, we conclude that the MSIV-LCS proposed by the licensees generally complies with the design guidelines of Regulatory Guide 1.96, and is, therefore, acceptable for the DAEC site. We further conclude that the limiting conditions for operation and surveillance requirements proposed by the licensees will provide reasonable assurance that the MSIV-LCS will operate as designed.

2. Inservice Inspection Requirements for Class 1 and Class 2 Components

We have reviewed Amendment 16 to the Duane Arnold FSAR which revises FSAR Appendix J to include additional inservice inspection of Class 2 components and also revises portions of the inservice inspection program for Class 1 components. Section XI through the Winter 1972 Addenda has been used as a guide in the preparation of the DAEC Inservice Inspection plan for Class 1 and Class 2 components. All examinations will be conducted in accordance with the ASME Code, Section III, Appendix IX as required by Section XI, subparagraph IS-210.

Revised FSAR Appendix J contains tables for Class 1 and 2 components similar to Section XI, Tables IS-261 and ISC-261 describing the component examination method, inspection schedule and degree of conformance. Presently known instances where radiation levels, plant design and/or materials that make compliance with Section XI impractical are discussed. No examination will be performed which requires draining the reactor vessel or removal of the core solely for the purpose of inservice inspection.

Interior surfaces and components below the reactor core are not made accessible by normal refueling operations. The reactor core internals in this region will be visually examined when maintenance operations provide access. The internal pressure boundary surface of one of the two recirculation pumps will be visually inspected as required by the Code only if a pump is disassembled for maintenance since draining the reactor vessel or removal of the core is required. The internal pressure boundary surface of one of each Class 1 valve two inches in nominal size and over will be visually inspected only if the valve is disassembled for maintenance or other purposes.

The upgraded inservice inspection program for the Duane Arnold Energy Center as defined in Amendment No. 16 has been evaluated and found to be in compliance with the ASME Code, Section XI through the Winter 1972 Addenda to the maximum extent practical. Since the engineering and design effort associated with DAEC predates Section XI, access for specific inservice inspection is limited by physical obstructions or anticipated high radiation fields. To implement the total upgraded inservice inspection program IELD intends to develop automated or mechanized inspection systems.

The technical approach of accomplishing the required visual examination during refueling or maintenance is consistent with commercial practice for the following examination categories:

1. Category N (Table IS-261) Reactor Vessel Interior Surfaces and Internals and Integrally-Welded Internal Supports.
2. Category L-2 (Table IS-261) Pump Casings.
3. Category M-2 (Table IS-261) Valve Bodies.
4. Category C-H (Table ISC-261) Pump Casings.
5. Category C-H (Table ICS-261) Valve Bodies.

During the course of our review, we requested and the licensees agreed to submit a report to us at the end of each ten-year inspection interval defining the examination categories that could not be completed, specifically categories N, L-2 and M-2 for Class 1 components and Category C-H for Class 2 components.

We conclude that the licensees' inservice inspection program for Class 1 and Class 2 components is acceptable. We further conclude that the proposed Technical Specifications provide an acceptable implementation of the program.

3. Closure Times for Drywell and Torus Purge and Supply Valves

The current Technical Specifications indicate that the closure times for the purge supply, drywell purge outlet, and torus purge outlet power operated isolation valves must be in the range of 3 to 5 seconds. The licensee has proposed removing the lower limit on the closure times of these valves and instead, has proposed that the valves must close within five seconds. We agree with the lincesee that there is no undesirable transient that could occur if these valves closed in less than three seconds. Experience at other BWR plants indicates that these valves will not be damaged or otherwise adversely affected if they close in less than three seconds. Based on our review, we conclude that the maximum closure time of these valves should not exceed five seconds to assure that they would be closed before the onset of fuel failures following a postulated LOCA. We further conclude that the licensees' proposal to change the maximum closure time of these valves to five seconds is acceptable.

Environmental Considerations

We have determined that the 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), than an environmental statement, negative declaration, or environmental impact 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 change 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 change 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: March 17, 1976

TABLE 1

ESTIMATED CONSEQUENCES FROM A POSTULATED LOCA AT DUANE ARNOLD ENERGY CENTER

(ASSUMING FAILURE OF ONE INBOARD MSIV TO CLOSE)

	Doses, rem	
	Thyroid	Whole Body
0 - 2 hours @ EAB:		
Containment Leakage	13	2
MSIV-LCS Leakage	10	1
Total Dose	23	3
0 - 30 days @ LPZ:		
Containment Leakage	40	3
MSIV-LCS Leakage	233	1
Total Dose	273	4

TABLE 2

BASIC DATA USED TO ESTIMATE THE DOSE CONSEQUENCES FROM A POSTULATED LOCA AT DUANE ARNOLD ENERGY CENTER

Power Level	1658 Mwt
Fraction of Core Inventory Available in Drywell	
Iodines	25%
Noble Gases	100%
Distribution of Iodine Forms	
Elemental	91%
Organic	4%
Particulate	5%
Filter Efficiency	
Elemental	95%
Organic	95%
Particulate	99%
Drywell Free Volume	109,400 ft^3
Main Steam Line Isolation Valve Leak Rate	11.5 cfh/valve
Time of Systems Actuation	10 minutes
Delay Time in Line with Failed Valve	18.8 hours

X/Q Values, sec/m^3	Stack Release	Ground Level Release
0 - 0.5 hours @ EAB	9.5×10^{-5}	1.3×10^{-3}
0.5 - 2 hours @ EAB	1.7×10^{-5}	1.3×10^{-3}
0 - 8 hours @ LPZ	8.1×10^{-6}	2.2×10^{-5}
8 - 24 hours @ LPZ	2.5×10^{-6}	1.4×10^{-5}
24 - 96 hours @ LPZ	8.6×10^{-7}	5.1×10^{-6}
96 - 720 hours @ LPZ	2.4×10^{-7}	1.3×10^{-6}

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 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 consists of changes in the Technical Specifications that will (1) add limiting conditions for operation and surveillance requirements for a new main steam line isolation valve leakage control system, (2) update the inservice inspection requirements for Class 1 and Class 2 components, and (3) remove the lower limit for closure times of the purge supply, drywell purge outlet, and torus purge outlet power operated isolation valves.

The application for the amendment 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 amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will

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not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or 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 December 22, 1975, November 20, 1975 and February 24, 1976, (2) Amendment No. 18 to License No. 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. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 17 day of March 1976

FOR THE NUCLEAR REGULATORY COMMISSION

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

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