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. 169
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, et al., dated July 27, 1988, as revised June 29, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 169, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: September 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

3.2-20

3.2-45

3.6-5

3.6-8

3.6-9

3.6-25

3.6-26

3.6-36

3.6-37

TABLE 3.2-E

INSTRUMENTATION THAT MONITORS DRYWELL LEAK DETECTION

Minimum No. of Operable Instrument Channels	Instrument	No. of Instrument Channels Provided by Design	Action
1	Sump System (1)	6	(3)
1	Air Sampling System (2)	6	(3)

NOTES FOR TABLE 3.2-E

(1) The Sump System is comprised of the Equipment Drain Sump and Floor Drain Sump Sub-systems.

The Equipment Drain Sump Sub-system consists of one Equipment Drain Sump Flow Integrator and two Equipment Drain Sump Flow Timers. The Floor Drain Sump Sub-system likewise consists of one Floor Drain Sump Flow Integrator and two Floor Drain Sump Flow Timers. The Sump Sub-system is operable when any one of these six devices operable.

(2) The Air Sampling System provides a backup system to the Sump System.

Action for Table 3.2-E

(3) See Specification 3.6.C.

timer is set to annunciate before the values specified in Specification 3.6.C are exceeded. An air sampling system is also provided, as a backup to the sump system, to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

On July 26, 1984 the NRC published their final rule on Anticipated Transients Without Scram (ATWS), (10 CFR § 50.62). This rule requires all BWR's to make certain plant modifications to mitigate the consequences of the unlikely occurrence of a failure to scram during an anticipated operational transient. The bases for these modifications are described in NEDE-31096-P-A, "Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.62," December 1985. The Standby Liquid Control System (SLCS) was modified for two-pump operation to provide the minimum required flowrate and boron concentration required by the ATWS rule (see section 3.4 Bases). The existing ATWS Recirculation Pump Trip (RPT) was modified from a one-out-of-two-once logic to trip each recirc. pump to a two-out-of-two-once logic to trip both recirc. pumps ("Monticello" design). This logic will also initiate the Alternate Rod Insertion (ARI) system, which actuates solenoid valves that bleed the air off the scram air header, causing the control rods to insert. The instrument setpoints are chosen such that the normal reactor protection system (RPS) scram setpoints for reactor high pressure or low water level will be exceeded before the ATWS RPT/ARI setpoints are reached. Because ATWS is considered a very low probability event and is outside the normal design basis for the DAEC, the surveillance frequencies and LCO requirements are less stringent than for safety-related instrumentation.

The End-of-Cycle (EOC) recirculation pump trip was added to the plant to improve the operating margin to fuel thermal limits, in particular Minimum Critical Power Ratio (MCPR). The EOC-RPT trips the recirc. pumps to lessen the severity of the power increases caused by either a closure of turbine

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Coolant Leakage	C. Coolant Leakage
1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:	1. Reactor coolant system leakage shall be checked by the sump system and recorded at least once every 8 hours.
a. 5 gpm unidentified leakage.	2. The air sampling system shall be checked and recorded at least once every 8 hours.
b. 2 gpm increase in unidentified leakage within a 24 hr. period.	
c. 25 gpm total leakage.	
2. The sump system shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. From and after the date that the sump system is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 24 hours unless the system is made operable sooner, provided the air sampling system is operable.	
3. If the conditions in 1 or 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	
D. Safety and Relief Valves	D. Safety and Relief Valves
1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, both safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.	1. At least one safety valve and 3 relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles. The setpoint of the safety valves shall be as specified in Specification 2.2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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	G. Structural Integrity
The structural integrity of the pressure boundaries shall be maintained at the level required by the original acceptance standard throughout the life of the plant.	1. In-service inspection of ASME Code Class I, Class II and Class III components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
	a. The second 10 year interval for the inservice inspection program described above commenced on November 1, 1985.
	2. In-service testing of ASME Code Class I, Class II and Class III pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
	a. The second 10-year interval for the inservice testing program described above commenced on February 1, 1985.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	3. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in this generic letter.

establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

Identified and unidentified leakage are defined in the DAEC Updated FSAR, Section 5.2.5.2.2. Total leakage is defined as the sum of identified and unidentified leakage.

The capacity of the drywell floor sump pumps is 50 gpm and the capacity of the drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin.

DAEC surveillance procedures require both identified and unidentified leakage to be determined at approximately 4 hour intervals. Should leakage exceed the allowed limits, control room alarms actuated by the equipment drain sump and floor drain sump pump timers are provided to indicate this condition, thus, continuous leakage detection capability is provided by design.

The requirement that an increase in unidentified leakage shall not exceed 2 gpm in a 24 hour period is based on maintaining the ability to detect small leaks in a reasonably short time such that corrective action can be initiated. However, during reactor startup and ascension to normal operating pressure, leakage should be closely monitored until normal operating pressure is achieved and a "baseline" leakage rate can be established to which any leakage increase can be compared.

The primary containment atmosphere radioactivity detector provides a sensitive and rapid indication of increased nuclear system leakage. The primary containment environmen[illegible]uously sampled from one of three locations which are chosen to [illegible] both a representative gas mixture and an indication of the location of the leakage.

The sample air undergoes three separate processes in which the radioactive noble gas, halogen, and particulate contents are determined. This system is thus a three channel monitoring system. The processed air is returned to the drywell.

The primary containment atmosphere radioactivity detector serves as a sensitive, reliable backup to the other methods of leak detection. It is anticipated that the particulate detector will be the primary indication of leakage, with the halogen and noble gas detectors serving as indication of the primary containment environment if primary containment venting is required. These detectors in conjunction with an isotopic analysis can be used to indicate whether the detected leak is from a steam or water system. This system is not capable of accurately quantifying coolant leakage rates. Because the Air Sampling system is not capable of determining leak rate, it is considered a backup system to the sump system, and no LCO is associated with it. It is intended to be a compensatory measure used when the sump system is inoperable.

The first 10-year interval for inservice inspections in accordance with the ASME Boiler and Pressure Vessel Code, Section XI commenced on February 1, 1975. This interval was extended for 9 months because of a 9 month outage for replacement of recirculation system inlet nozzle safe-ends in 1978-79. Therefore, the first 10-year interval ended on October 31, 1985.

The second 10-year interval for inservice inspections commenced on November 1, 1985 and is scheduled to end on October 31, 1995. The second 10-year inspection program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications as stated in 10 CFR 50.55a.

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 sponsered 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. Section 5.2.4 of the Updated FSAR provides details of the inservice inspection program.

Starting with the Cycle 9/10 Refueling Outage, an augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC staff's positions set forth in Generic Letter 88-01 and NUREG-0313, Revision 2 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

The first 10-year interval for inservice testing of pumps and valves in accordance with the ASME Code, Section XI commenced on February 1, 1975 and ended on January 31, 1985. The second 10-year inservice testing interval commenced on February 1, 1985 and is scheduled to end on January 31, 1995. The second 10-year testing program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications of 10 CFR 50.55a. Section 3.9.6 of the Updated FSAR describes the inservice testing program.

3.6.H & 4.6.H BASES:

<u>Shock Suppressors (Snubbers)</u>

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or other severe transient, while accommodating normal thermal motion during system startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of damage to piping as a result of a seismic or other event initiating dynamic loads or, in the case of a frozen snubber, exceeding allowable stress limits during system thermal transients. 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.