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DESCRIPTION	ENCLOSURE
Notorized 1/10/78 trans the following:	License No. DPR-49 Appl for Amend: tech specs proposed change concerning the Automatic Depressurization System and surveillance requirements & Bases for Safety & Relief Valves and Limiting Safety System Settings for Safety & Relief Valves....
PLANT NAME: Duane Arnold RJL 1/16/78 (1-P)	(1-P)+(14-P) 40 ENCL

FOR ACTION/INFORMATION

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January 10, 1978
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LEE LIU
VICE PRESIDENT - ENGINEERING

RECEIVED JAN 16 1978 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

50-331

Mr. Edson G. Case
Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Case:

Transmitted herewith, in accordance with the requirements of 10 CFR 50.59 and 50.90, is an application for amendment of DPR-49 (Appendix A to License) for the Duane Arnold Energy Center as requested by the Nuclear Regulatory Commission letter of December 8, 1977.

This application, consisting of proposed Technical Specification change RTS-104 has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee. This application does not involve a significant hazards consideration.

Three signed and notarized originals and 37 additional copies of this application are transmitted herewith. This application, consisting of the foregoing letter and enclosures hereto, is true and accurate to the best of my knowledge and belief.

Iowa Electric Light and Power Company

By [signature]
Lee Liu
Vice President-Engineering

LL/OCS/D
Encs.
cc: Mr. D. Arnold
Mr. K. Meyer
Mr. R. Lowenstein
Mr. J. Keppler (NRC)
Mr. R. Clark (NRC)
Mr. L. Root
File: A-117

Subscribed and sworn to before me this 10th day of January, 1978.

Sue E. Scott
Notary Public in and for the State of Iowa.

NOTARIAL SEAL IOWA
SUE E. SCOTT
My Commission Expires
September 30, 1978

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PROPOSED CHANGE RTS-104 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix A of the Technical Specifications for the DAEC (DPR-49) provides as follows:

Specifications 3.5.F and 4.5.F contain Limiting Conditions for Operation, Surveillance Requirements and Bases for the Automatic Depressurization System and Specifications 3.6.D and 4.6.D contain Limiting Conditions for Operation, Surveillance Requirements and Bases for Safety and Relief Valves. In addition, Specifications 2.2.1.B and 2.2.1.C contain Limiting Safety System Settings for Safety and Relief Valves.

II. Proposed Changes in Technical Specifications

The licensees of DPR-49 propose the following changes in the Technical Specifications set forth in I above:

Delete pages vi, 1.2-1, 1.2-2, 3.5-9, 3.5-21, 3.5-22, 3.6-5, 3.6-6 and 3.6-23 through 3.6-25, and replace with the attached sheets.

III. Justification for Proposed Change

This change is proposed in response to a request from the Nuclear Regulatory Commission (Letter; Mr. G. Lear, Chief, Operating Reactors Branch #3, Division of Operating Reactors, to Mr. D. Arnold, President, Iowa Electric Light and Power Company, December 8, 1977).

IV. Review Procedure

This proposed change has been reviewed by the DAEC Operations Committee and Safety Committee which have found that this proposed change does not involve a significant hazards consideration.

Table Number	Title	Page
4.2-D	Minimum Test and Calibration Frequency for Radiation Monitoring Systems	3.2-29
4.2-E	Minimum Test and Calibration Frequency for Drywell Leak Detection	3.2-30
4.2-F	Minimum Test and Calibration Frequency for Surveillance Instrumentation	3.2-31
4.2-G	Minimum Test and Calibration Frequency for Recirculation Pump Trip	3.2-34
3.6-1	Number of Specimens by Source	3.6-33
4.6-1	Nuclear Class I Access Provisions and Examination Schedule	3.6-34
4.6-2	Nuclear Class II Access Provisions and Examination Schedule	3.6-39
4.6-3	Snubbers Accessible During Normal Operation	3.6-41
4.6-4	Snubbers Inaccessible During Normal Operation	3.6-43
4.6-5	Snubbers in High Radiation Area During Shutdown and/or Especially Difficult to Remove	3.6-44
4.6-6	Remotely Operated Relief and Safety Relief Valve Test Schedule	3.6-46
3.7-1	Containment Penetrations Subject to Type "B" Test Requirements	3.7-20
3.7-2	Containment Isolation Valves Subject to Type "C" Test Requirements	3.7-22
3.7-3	Primary Containment Power Operated Isolation Valves	3.7-25
3.12-1	Significant Input Parameters to the Duane Arnold Loss-of-Coolant Accident Analysis	3.12-9
3.12-2	MCPR Limits	3.12-9a
6.2-1	Minimum Shift Crew Personnel and License Requirements	6.2-3
6.9-1	Protection Factors for Respirators	6.9-8
6.11-1	Reporting Summary - Routine Reports	6.11-12
6.11-2	Reporting Summary - Non-routine Reports	6.11-14

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

1. The reactor vessel dome pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification:

1. The limiting safety system settings shall be as specified below:

Protective Action/Limiting Safety System Setting

A. Scram on Reactor Vessel high pressure

1035 psig

B. Relief valve settings

1090 psig ± 1% (2 valves)
1100 psig ± 1% (2 valves)
1110 psig ± 1% (2 valves)

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	C. Safety Valve settings 1240 psig ± 1% (2 valves)
2. The reactor vessel dome pressure shall not exceed 135 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.	2. The shutdown cooling isolation valves shall be closed whenever the reactor vessel dome pressure is $\geq$ 135 psig.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

F. Automatic Depressurization System (ADS)

1. The ADS shall be operable with at least three operable ADS valves whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 100 psig and prior to a startup from a Cold Condition, except as specified in 3.5.F.2 and 3.5.F.3.

2. With one of the above required ADS valves inoperable, operation may continue provided the actuation logic of the remaining ADS valves is operable and the Core Spray and LPCI Systems are operable, and the HPCI System is demonstrated operable within four hours; restore the inoperable ADS valve to operable status within 14 days or be in at least Hot Shutdown within the following 12 hours and in Cold Shutdown within the following 24 hours.

3. With two or more of the above required ADS valves inoperable, be in at least Hot Shutdown within 12 hours and in Cold Shutdown within the following 24 hours.

SURVEILLANCE REQUIREMENTS

F. Automatic Depressurization System (ADS)

1. In addition to the applicable ASME Boiler and Pressure Vessel Code, Section XI requirements, the ADS shall be demonstrated operable:

a. At least once per 18 months by the performance of a system functional test which includes simulated automatic actuation through the automatic depressurization sequence, but excluding valve actuation.

b. Until March 1, 1979, at least once per 18 months by:

(1). Manually opening each ADS valve with a reactor steam dome pressure $\geq$ 100 psig, and verifying each valve opens by observing that either:

(a). The turbine bypass or control valve(s) indicate a compensating valve movement, or

(b). The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

(2). Conducting a visual inspection of the safety relief and relief valve line restraints in the torus to verify structural integrity for continued operation.

c. After March 1, 1979, by performance of the following test program:

(1). Manually opening each ADS valve in accordance with the test schedule of Table 4.6-6 with reactor steam dome pressure $\geq$ 100 psig, and verifying each valve opens by observing either:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	(a). The turbine bypass or control valve(s) indicate a compensating valve movement, or
	(b). The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.
	(2). The initial Next Required Test Interval of Table 4.6-6 shall be determined by the number of remotely operated relief and safety relief valves found inoperable from March 1, 1978 to March 1, 1979.
	(3). The initial valve tests of Table 4.6-6 shall be completed by, the earlier of:
	(a). The completion of the next refueling outage occurring after March 1, 1979, or
	(b). The time period defined by March 1, 1979, plus the initial test interval, determined above.
	4. At least once per 18 months by conducting a visual inspection of the safety relief and relief valve line restraints in the torus to verify structural integrity for continued operation.
G. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>	G. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>
1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided	1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In

E. RCIC System

The RCIC is designed to provide makeup to the nuclear system as part of the planned operation for periods when the main condenser is unavailable. RCIC also serves for decay heat removal when feedwater is lost. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgments on the reliability of the HPCI system, an allowable repair time of one month is specified. Immediate and weekly demonstrations of HPCI operability during RCIC outage is considered adequate based on judgment and practicality.

F. Automatic Depressurization System (ADS)

Upon failure of the HPCIS to function properly after a small break loss-of-coolant accident, the ADS automatically causes the safety relief valves to open, depressurizing the reactor so that flow from the low pressure cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be operable whenever reactor vessel pressure exceeds 150 psig even though low pressure cooling systems provide adequate core cooling up to 450 psig.

ADS automatically controls four safety relief valves although the safety analysis only takes credit for three. Therefore, only three ADS valves are required to be OPERABLE.

The testing frequency applicable to ADS valves is provided to ensure operability and demonstrate reliability of the valves. The required testing interval varies with observed valve failures. The number of inoperable valves found during both operation and testing of these valves determines the time interval for the next required test of these valves. Early tests may be performed prior to entering the next required test interval (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same time interval, however, they are not acceptable for lengthening the test interval since they were not performed within the ± 25% tolerance band as required by Table 4.6-6.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
C. <u>Coolant Leakage</u>	C. <u>Coolant Leakage</u>
1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212^{o}F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per day.
2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding seven days unless system is made operable sooner.	
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.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION

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, at least the following reactor coolant system safety valves and safety relief valves shall be operable with lift settings within ± 1% of the indicated pressures, except as specified in 3.6.D.2 and 3.6.D.3:

   2 Safety valves @ 1240 psig
   2 Safety relief valves @ 1110 psig
   2 Safety relief valves @ 1100 psig
   2 Safety relief valves @ 1090 psig

2. With the safety valve function of one of the above required safety relief valves inoperable, continued reactor operation is permissible only during the succeeding 30 days unless such valve function is sooner made operable.

3. With one or more of the above required reactor coolant system safety valves or the safety function of two or more of the above required safety relief valves inoperable, either restore the inoperable valve(s) to operable status or be in at least Hot Shutdown within 12 hours and Cold Shutdown within the following 24 hours.

SURVEILLANCE REQUIREMENTS

D. Safety and Relief Valves

1. In addition to the applicable ASME Boiler and Pressure Vessel Code, Section XI requirements, each safety relief valve shall be demonstrated operable:

a. At least once per 24 hours, by verifying bellows integrity through instrument indication.

b. Until March 1, 1979, at least once per 18 months by:

(1). Manually opening each remotely operated safety relief valve with reactor steam dome pressure ≥ 100 psig, and verifying each valve opens by observing that either:

(a). The turbine bypass or control valve(s) indicate a compensating valve movement, or

(b). The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

(2). Conducting a visual inspection of the safety relief valve line restraints in the torus to verify structural integrity for continued operation.

c. After March 1, 1979, by performance of the following test program:

(1). Manually opening each remotely operated safety relief valve in accordance with the test schedule of Table 4.6-6 with reactor steam dome pressure ≥ 100 psig, and verifying each valve opens by observing that either:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	(a). The turbine bypass or control valve(s) indicate a compensating valve movement, or
	(b). The reactor coolant pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.
	(2). The initial Next Required Test Interval of Table 4.6-6 shall be determined by the number of remotely operated relief and safety relief valves found inoperable from March 1, 1978 to March 1, 1979.
	(3). The initial valve tests of Table 4.6-6 shall be completed by, the earlier of:
	(a). The completion of the next refueling outage occurring after March 1, 1979, or
	(b). The time period defined by March 1, 1979, plus the initial test interval, determined above.
	(4). At least once per 18 months, by conducting a visual inspection of the safety relief valve line restraints in the torus to verify structural integrity for continued operation.
	2. Each safety valve and the safety valve function of each safety relief valve shall be demonstrated operable per the requirements of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

TABLE 4.6-6

REMOTELY OPERATED RELIEF AND SAFETY RELIEF VALVE TEST SCHEDULE#

NUMBER OF REMOTELY OPERATED RELIEF AND SAFETY RELIEF VALVES FOUND INOPERABLE DURING OPERATION, TESTING OR TEST INTERVAL**	NEXT REQUIRED TEST INTERVAL*
0	18 Months ± 25%
1	184 Days ± 25%
2	92 Days ± 25%
≥ 3	31 Days ± 25%

* The required test interval shall not be lengthened more than one step at a time. Early tests may be performed prior to entering the "next required test interval" (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same interval, however, they are not acceptable for lengthening the test interval.

**Setpoint drift is not considered to be a valve failure for the purposes of this test schedule.

\# Each affected remotely operated relief and safety relief valve shall be demonstrated OPERABLE pursuant to Specifications 4.6.D.1.b(1), 4.6.D.1.c(1), 4.5.F.1.b(1) and 4.5.F.1.c(1), as applicable, within 36 hours after exceeding 100 psig, whenever maintenance, repair or replacement work is performed on a valve or its associated actuator. Successful tests performed under this provision may be used to satisfy the test requirements for a "required test interval" provided such tests are performed within the current "required test interval" and its associated tolerance band. Valve failures detected during testing under this provision shall not be considered inoperable valves for the purpose of this table.

3.6.D & 4.6.D BASES:

Safety Valves and Safety Relief Valves

The reactor coolant system safety valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1335 psig in accordance with the ASME Code. Each spring loaded safety valve is designed to relieve 642,000 lbs. per hour at the valve setpoint. The capacity of the relief/safety relief valves is designed to meet the SAR stated requirement that these valves shall function to prevent opening of the spring loaded safety valves. The spring loaded safety valves are not expected to be required to function under the most limiting transient, assuming proper relief/safety relief valve operation.

The analysis of the worst overpressure transient (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in greater than a 79 psi margin to the code allowable overpressure limit of 1375 psig if a flux scram is assumed. In addition, the generic analyses have been conducted which show an approximate 36 psi sensitivity increase for each relief valve failure.

The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraphs 14.5.1.2 and 14.5.1.3 and is evaluated in each reload analyses. These analyses show that the six relief valves assure greater than 36 psi margin below the setting of the safety valves. Therefore, the safety valves will not open. These analyses verify that peak system pressure is limited to greater than a 125 psi margin to the allowed vessel overpressure of 1375 psig.

The testing frequency applicable to the relief valve function of the safety relief valves is provided to ensure operability and demonstrate reliability of the valves. This variable frequency test schedule becomes effective on March 1, 1979. This lead time is intended to permit resolution of the Mark I Safety Relief Valve Loads and Structural Capability generic concern. The required testing interval varies with observed valve failures. The number of inoperable valves found during both operation and testing of these valves determines the time interval for the next required test of these valves. Early tests may be performed prior to entering the next required test interval (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same time interval; however, they are not acceptable for lengthening the test interval since they were not performed within the ± 25% tolerance band as required by Table 4.6-6.

Demonstration of the safety valves and safety relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.