July 7, 1988

Docket No. 50-331

Mr. Lee Liu
Chairman of the board and
Chief Executive Officer
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Dear Mr. Liu

SUBJECT: AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-49: ATWS MODIFICATIONS TO COMPLY WITH 10 CFR 50.62 (TAC NO. 64789)

The Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your submittal dated February 26, 1987, as supplemented June 1, July 10, and November 13, 1987.

The amendment revises the DAEC Technical Specifications to reflect modifications made to comply with 10 CFR 50.62, the ATWS rule. These modifications consisted of the addition of an Alternate Rod Injection System, improvements to the Recirculation Pump Trip logic circuitry, and changes to the Standby Liquid Control System pump control circuitry. Compliance with the requirements of 10 CFR 50.62 was confirmed in NRC Inspection Report No. 50-331/87-25, issued on October 5, 1987.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/S/

James R. Hall, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 151 to
   License No. DPR-49
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

*SEE PREVIOUS CONCURRENCE

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 151
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 February 26, 1987, as supplemented June 1, 1987, 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. 151 , 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

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 7, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 151

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.

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LIMITING CONDITION FOR OPERATION

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-E.

F. Surveillance Information Readouts

The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2-F.

G. Recirculation Pump Trips (RPT) and Alternate Rod Insertion (ARI)

(ATWS) - RPT/ARI

The limiting conditions for operation for the instrumentation that trips the recirculation pumps and initiates Alternate Rod Insertion (ARI) as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-G.

(EOC) - RPT

The limiting conditions for operation for the instrumentation that trips the recirculation pumps during turbine stop valve or control valve fast closure for transient margin improvement (especially for end of cycle) are given in Table 3.2-G.

H. Accident Monitoring Instrumentation

The limiting conditions for operation for the accident monitoring instrumentation are given in Table 3.2-H.

SURVEILLANCE REQUIREMENT

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2-E.

F. Surveillance Information Readouts

Instrumentation shall be calibrated and checked as indicated in Table 4.2-F.

G. Recirculation Pump Trips and Alternate Rod Insertion

Instrumentation and logic shall be functionally tested, calibrated, and response time tested as indicated on Table 4.2-G.

H. Accident Monitoring Instrumentation

Instrumentation shall be calibrated and checked as indicated in Table 4.2-H in all operational modes other than COLD SHUTDOWN or refueling.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP (RPT) AND/OR ALTERNATE ROD INSERTION (ARI)

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action
2	(ATWS) RPT/ARI Reactor High Pressure	$\leq$ 1140 psig	4*	(2) (6)
2	(ATWS) RPT/ARI Reactor Low-Low Water Level	$\geq$ +119.5 in. Indicated level(5)	4*	(2) (6)
1	(EOC) RPT Logic	N/A	2	(3)
1	(EOC) RPT System (Response Time)	$\leq$ 140 msec (4)	2	(3)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter. If this cannot be met, the indicated action shall be taken.

2. With one instrument channel inoperable, restore the inoperable instrument channel to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 24 hours. With both instrument channels inoperable, restore at least one instrument channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 24 hours.

3. Two EOC RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 consecutive hours, an orderly power reduction shall be initiated and the reactor power shall be less than 85% within four hours.

4. This response time is from initiation of Turbine control valve fast closure or Turbine stop valve closure to actuation of the breaker secondary (auxiliary) contact.

5. Zero referenced to top of active fuel.**

6. If an instrument(s) is inoperable, it may be considered to be OPERABLE if placed in a tripped condition.

*There are 2 instruments per trip system that are arranged in a two-out-of-two once logic.

**Top of active fuel zone is defined to be 344.5" above vessel zero (see bases 3.2).

TABLE 4.2-G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP (RPT) AND/OR ALTERNATE ROD INSERTION (ARI)

(ATWS - RPT/ARI)

Instrument Channel	Instrument Functional Check	Calibration Frequency
Reactor High Pressure	Annual	Annual
Reactor Low Water Level	Annual	Annual

System Function Test	Frequency
Recirculation Pump Trip	Once/operating cycle
Alternate Rod Insertion	Once/operating cycle

(EOC - RPT)

Instrument Channel	Functional Check	Calibration Frequency	Instrument Check	Response Time
RPT Initiate Logic	Once/Month	N/A	N/A	N/A
RPT Breaker	Once/Operating Cycle	N/A	N/A	Once/Operating Cycle

timer is set to annunciate before the values specified in Specification 3.6.C are exceeded. An air sampling system is also provided 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 stop valves or fast closure of the turbine control valves with reactor power greater than 30% and a simultaneous failure of the turbine bypass valves to open. The operating limit MCPR of section 3.12.C is calculated assuming an operable EOC-RPT system. If the requirements of Table 3.2-G are not met, then the reactor power level is reduced to a level (85% of rated) which will ensure that the full-power MCPR limits of section 3.12.C will not be violated if such a transient were to occur.

Trip function settings are included for instrument a.c. and battery busses for surveillance of undervoltage relays. The undervoltage relays are required to sense a reduction in the power source voltage so that the subject instruments can be transferred to an alternate power source.

Surveillance tests other than a monthly functional check of the bus power monitors for the RHR, Core Spray, ADS, HPCI and RCIC trip systems are not required since they serve as annunciators for complete loss of power and do not monitor reduction of voltage. The subject functional check consists of opening the appropriate circuit breakers and observing the loss of power annunciator activation.

The accident monitoring instrumentation listed in Table 3.2-H were specifically added to comply with the requirements of NUREG-0737 and Generic Letter 83-36. The instrumentation listed is designed to provide plant status for accidents that exceed the design basis accidents discussed in Chapter 15 of the DAEC UFSAR.

Footnote 9 of Table 3.2-H deviates from the guidance of Generic Letter 83-36 as continued operation for 30 days (instead of 7 days as recommended in the generic letter) is allowed with one of two torus water level monitor (TWLM) channels inoperable. Continued operation is justified by the following considerations:

1) Redundancy is available in that at least one channel of the containment water level monitor (CWLM) instrumentation must be available. Since the CWLM envelopes the span measured by the TWLM, the torus water level can be monitored by the CWLM system.

3.4 BASES

Standby Liquid Control System

1. The purpose of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration of 600 ppm of boron in the reactor core in less than 96 minutes based on a minimum 26.2 gpm pump rate. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, analytical biases and uncertainties, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

   The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

2. Although the standby liquid control pump circuitry has been modified to run both pumps simultaneously in order to comply with the ATWS

rule requirements,* only one of the two standby liquid control pumps is needed for meeting the SLCS design basis. One inoperable pumping circuit does not immediately threaten shutdown capability, and reactor operation can continue while the circuit is being repaired. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained for a one-out-of-two system by an allowable equipment out-of-service time of one third of the normal surveillance frequency. This method determines an equipment out-of-service time of ten days. Additional conservatism is introduced by reducing the allowable out-of-service time to seven days, and by increased testing of the operable redundant component.

3. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration range. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

---

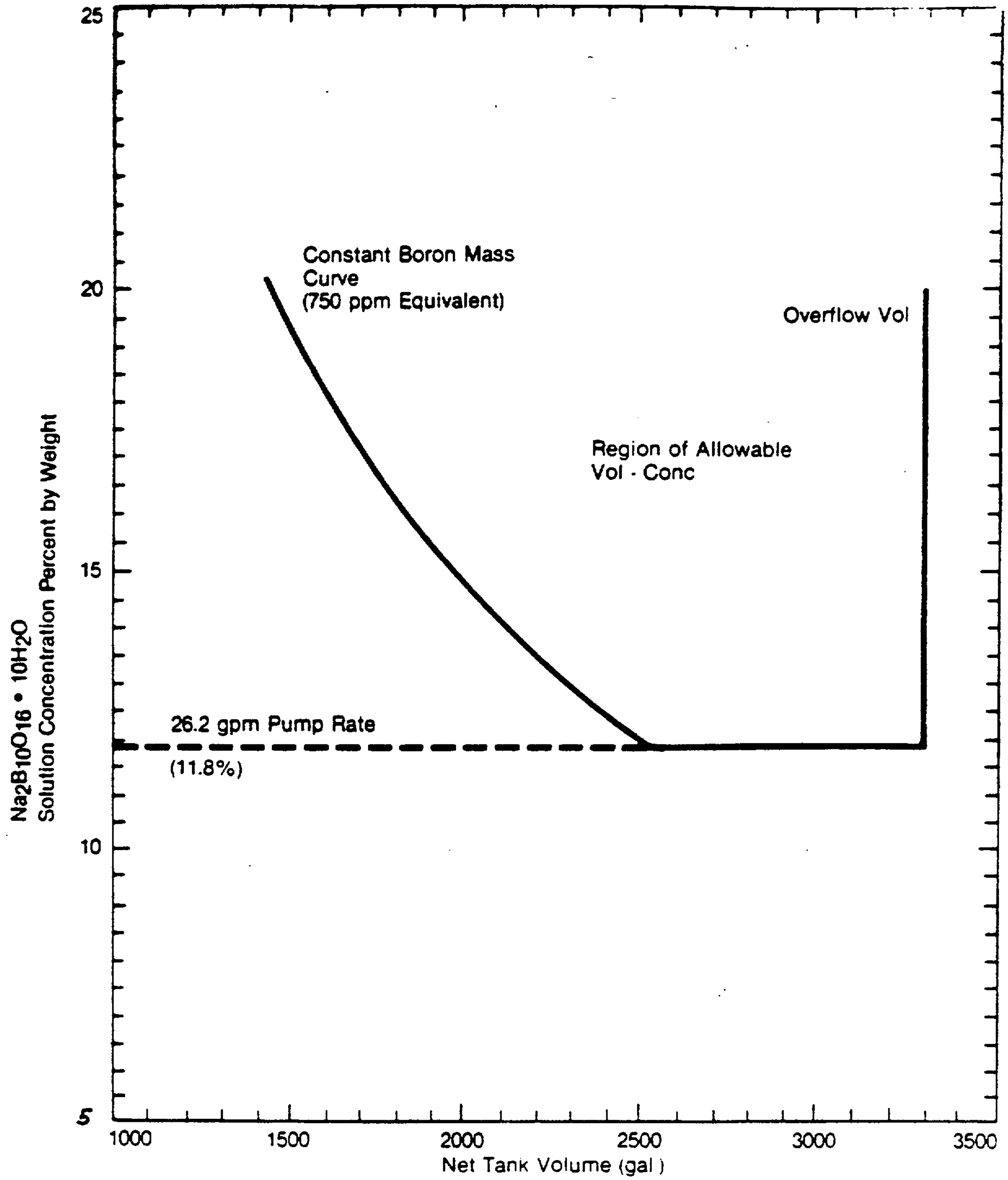
* The NRC's final rule on Anticipated Transients Without Scram (ATWS), 10 CFR § 50.62, requires that the Standby Liquid Control System (SLCS) be modified to provide an equivalent shutdown capability of 86 gpm at 13.4 weight percent natural boron for a 251 inch I.D. vessel. For the DAEC, ATWS equivalence is achieved by running both SLCS pumps simultaneously at a minimum combined flow of 45 gpm at a nominal boron concentration of 13% weight percent natural boron, (NEDC-30859, "Duane Arnold ATWS Assessment," December 1984). (The equivalence is also met if both pumps supply their minimum tech spec flowrate of 26.2 gpm each with a solution concentration of at least 11.2 weight percent natural boron.) Because ATWS is a very low probability event and is considered to be beyond the design basis for the DAEC, the surveillance and LCO requirements need not be more stringent than the original SLCS design basis requirements.

The solution is kept at least 5°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4-2.

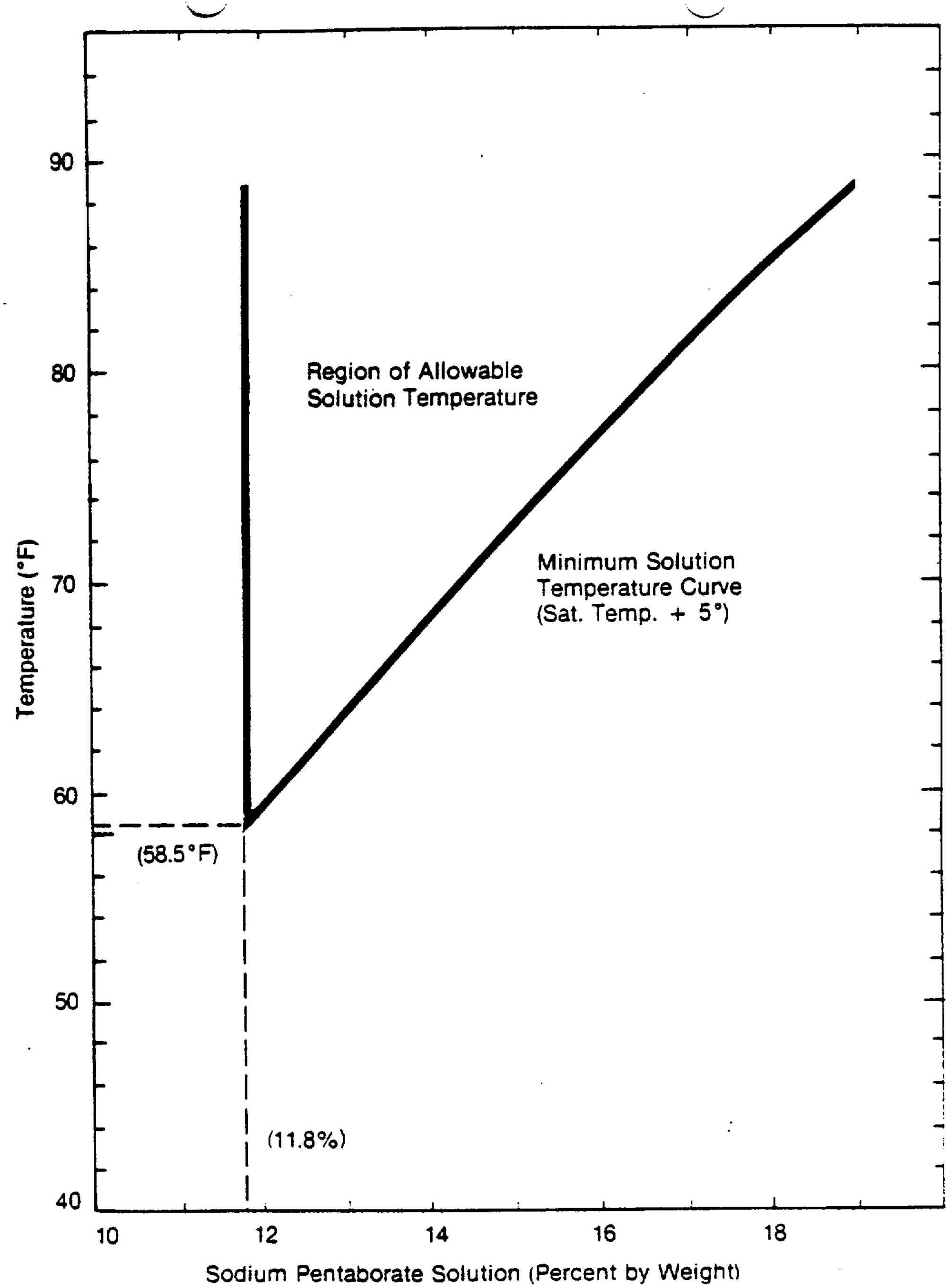
The volume versus concentration requirement of the solution is such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature versus concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 2500 gallons of solution having a 11.8 percent sodium pentaborate concentration, or the equivalent as shown in Figure 3.4-1, will ensure that both the SLCS design basis (11.8%) and ATWS (11.2%) shutdown requirements are met. The maximum net storage volume of the boron solution, as established by the overflow, is 3270 gallons.



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY TECHNICAL SPECIFICATIONS
Sodium Pentaborate Solution Volume Concentration Requirements FIGURE 3.4-1



DUANE ARNOLD ENERGY CENTER IOWA ELECTRIC LIGHT & POWER COMPANY TECHNICAL SPECIFICATIONS
Minimum Temperature of Sodium Pentaborate Solution FIGURE 3.4-2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-49

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

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

On July 26, 1984, Title 10 of the Code of Federal Regulations (CFR) was amended to include Section 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (know as the "ATWS Rule"). An ATWS is an expected operational transient (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) which is accompanied by a failure of the reactor trip system (RTS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients, and to mitigate the consequences of an ATWS event.

For each boiling water reactor, three systems are required to mitigate the consequences of an ATWS event.

1. It must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation devices. The ARI system must have redundant scram air header exhaust valves. The ARI system must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

2. It must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection location must be designed to perform its function in a reliable manner.

3. It must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

By letters dated February 26, June 1, July 10 and November 13, 1987 (Refs. 1, 2, 3, & 4), Iowa Electric Light and Power Company (the licensee) provided information to comply with the ATWS Rule. This safety evaluation addresses the licensee's proposed implementation of the ATWS Rule requirements.

2.0 REVIEW CRITERIA

The systems and equipment required by 10 CFR 50.62 do not have to meet all of the stringent requirements normally applied to safety-related equipment. However, this equipment is part of the broader class of structures, systems, and components important to safety defined in the introduction to 10 CFR Part 50, Appendix A, General Design Criteria (GDC). GDC-1 requires that "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." Generic Letter 85-06, "Quality Assurance Guidance For ATWS Equipment That is Not Safety-Related," details the quality assurance that must be applied to this equipment.

In general, the equipment to be installed in accordance with the ATWS Rule is required to be diverse from the existing RTS, and must be testable at power. This equipment is intended to provide needed diversity (where only minimal diversity currently exists in the RTS) to reduce the potential for common mode failures that could result in an ATWS leading to unacceptable plant conditions.

The criteria used in evaluating the licensee's submittal include 10 CFR 50.62 "Considerations Regarding System and Equipment Criteria" published in Federal Register Volume 49, No. 124, dated June 26, 1984, and Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That is Not Safety-Related."

3.0 DUANE ARNOLD ARI & ATWS/RPT SYSTEM DESCRIPTION

The Duane Arnold Energy Center (DAEC) has installed a redundant ARI/RPT initiation logic system to mitigate the potential consequences of an anticipated transient without scram event. The system consists of reactor pressure and reactor water level sensors, logic, power supplies and instrumentation that is independent from the reactor trip system. It is a two divisional safety-related system. Each division is capable of initiating protective actions when both input channels (either pressure or level) within a division are tripped. The ARI/RPT system output will energize the devices to start the protective actions. The system can be manually initiated by depressing two pushbuttons simultaneously on the control panel. The ARI logic will cause the immediate energization of the ARI valves when either the reactor vessel high pressure trip setpoint or the low-low water level trip setpoint is reached.

The same logic will also trip the reactor coolant recirculation pumps. There are two breakers in series at each pump power feeder. Each logic train signal will trip one of the two breakers. Either logic train will trip both pumps.

The DAEC RPT logic delays recirculation pump trip on low-low water level for 9 seconds to allow the low-pressure coolant injection (LPCI) system loop selection logic to complete its function. This function is detection of recirculation line break and selection of the LPCI injection point. No time delay is provided on a high reactor pressure vessel (RPV) pressure signal.

4.0 EVALUATION

The licensee participated in the BWR Owners Group ATWS implementation alternatives program. The BWR Owners Group submitted a licensing topical report, NEDE-31096-P, "Anticipated Transients Without Scram, Response to NRC ATWS Rule 10 CFR 50.62" (Ref. 5) for staff review. The staff accepted the topical report in Reference 6. Reference 1 summarizes the licensee's compliance with the ATWS Rule. The staff's evaluation is addressed in the following sections.

ARI

In a letter dated June 1, 1987 (Ref. 2), the licensee stated that, based on in-plant test data, the rod motion actually begins at approximately 10 seconds for the first rod and 30 seconds for the last rod. All rod motions will be completed within 37 seconds. A post-test evaluation determined that a choked flow condition exists in the scram valves for the individual control rod drives and that modifications to the design of the ARI system would not improve ARI performance.

The licensee requested General Electric to evaluate the test data. General Electric evaluated these data and concluded that, although the observed 30-second time delay is larger than that used in Reference 5, the design objectives would still be met as long as rod motion was completed within 60 seconds. The ATWS licensing topical report also recognizes that an important reason for minimizing the rod motion completion time is to ensure that the scram discharge volumes (SDV's) have sufficient volume to accommodate whatever leakage will occur during the time when the air header is bleeding down and rod motion has not begun. The test results indicate that sufficient volume exists in the DAEC SDV's to allow all control rods to complete their motion. The staff has evaluated the licensee's justification and concluded that the ARI system function time at the DAEC is acceptable.

The ATWS Rule does not require the ARI system to be safety grade, but the implementation must be such that the existing protection system continues to meet all applicable safety-related criteria.

The licensee stated that the Duane Arnold (DAEC) ARI/RPT logic provides signals for Alternate Rod Injection and the recirculation pump trip. It is designed as a Class 1E system. The ARI/RPT logic is totally independent from the existing reactor protection system except for annunciators. Contact isolation is provided between the annunciator and the initiation circuitry. Any failure in the annunciator system will not cause an ARI/RPT logic failure or prevent the existing reactor trip system (RTS) from performing its protective functions. The staff finds this acceptable.

The licensee stated that the ARI system has redundant valves at the scram air header. The ARI system performs a function redundant to the backup scram system and the RPS. The staff finds this acceptable.

The licensee stated that the ARI system is diverse and independent from the reactor trip system up to the scram air header. The ARI solenoid valves are DC powered with the solenoid valves energized to open. The ARI solenoid valves are separated from the backup scram valves. All instrument channel components, including the sensors, will be diverse from the existing RTS components. The staff finds this acceptable.

The licensee stated that the ARI actuation logic is separated from the RTS logic. The ARI/RPT system is electrically independent from the existing RTS. The staff finds this acceptable.

The licensee stated that the ARI system is physically separated from the RTS. Wiring for the RTS outside of the enclosures in the control room is run in rigid metallic conduits. All RTS conduits are identified by an alpha-numeric designator. The ARI/RPT system wiring uses scheduled cables and raceways. The ARI/RPT cable routing does not violate the DAEC RTS channel separation criteria. DAEC Quality Control personnel verified that the cables were installed as designed. The staff finds this acceptable.

The licensee stated that all hardware required for the ARI system to function will be environmentally qualified to conditions that occur during an anticipated operational occurrence. The staff finds this acceptable.

The licensee has committed to comply with Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That is Not Safety-Related." The staff finds this acceptable.

The licensee stated that the ARI logic is classified as safety-related and is powered from divisional 125 Vdc batteries. The noninterruptible power sources allow the ARI to perform its intended function during any loss of offsite power event. Circuit breakers are provided at the dc distribution panels that protect the ARI/RPT feeders. The ARI/RPT logic trains and RPT breaker trip coils are individually fused. The staff finds this acceptable.

The licensee stated that the ARI system will be testable up to and including the final actuating devices while the reactor is at power. The ARI system is comprised of two identical logic trains. Each logic train is equipped with an ARI solenoid valve. During the ARI surveillance test, a mode switch is set to the test position, which will block the ARI solenoid valve from being energized on one train. The other train still can be initiated by the automatic signals. The mode switch in the test position will not affect the manual initiation capability of the ATWS system. The staff finds this acceptable.

The licensee stated that the ARI design will utilize coincident logic. Both channels must be tripped in order to initiate the mitigative actions. The ARI actuation setpoints will not challenge scram setpoints. The staff finds this acceptable.

The licensee stated that manual initiation capability will be provided. The staff finds this acceptable.

The licensee stated that indication will be provided to indicate when the system is actuated, in test, or out of service. Four annunciators will be provided. They are:

1. ATWS Channel A Initiated
2. ATWS Channel B Initiated
3. ATWS Channel A in Test
4. ATWS Channel B in Test

The staff finds this acceptable.

The licensee stated that the ARI design will have a seal-in feature to ensure the completion of protective action once it is initiated. After removal of the initiating signal, the logic will automatically reset after a preset time delay to allow manual scram. The staff finds this acceptable.

As stated in Reference 6, the staff SE on GE Topical Report NEDE-31096-P, the staff does not intend to repeat its review of the design information described in the GE Topical Report and found acceptable when the report appears as a reference in a specific license amendment application. Reference 1 summarizes the licensee's compliance with the ATWS Rule. The staff finds that the Duane Arnold ARI design is in general compliance with the ATWS Rule, 10 CFR 50.62 paragraph (c)(3). The proposed Technical Specification changes associated with the ARI system are, therefore, acceptable.

<u>STANDBY LIQUID CONTROL SYSTEM (SLCS)</u>

The ATWS Rule requires that the SLCS be equivalent in control capacity to a system with an 86 gpm injection rate, using 13 weight percent natural, unenriched sodium pentaborate solution, in a system with a 251-inch diameter reactor vessel. Of the several proposed approaches presented in the General Electric report (Ref. 5) and approved in the NRC safety evaluation (Ref. 6), the licensee has chosen to use the combined options of dual pump operation at 52.4 gpm and an increased minimum sodium pentaborate solution concentration from 9.8 to 11.8 weight percent.In Reference 3, the licensee calculated a minimum required concentration of 11.2 weight percent sodium pentaborate for an injection rate of 52.4 pgm, assuming a total water mass of 329,909 lbs. in the reactor vessel and associated piping. The approach taken for the DAEC is consistent with that approved by the staff in Reference 6 and the resulting parameters are, therefore, acceptable.

The changed values lead to proposed Technical Specification changes. These include doubling the pump flow rate from 26.2 gpm for single pump operation to 52.4 gpm for dual pump operation in TS 3.4.1 and changing Figure 3.4-1 for the SLC solution concentration versus tank volume curve. The changes are necessary to reflect the new concentration limits which will ensure that the plant meets both the new ATWS requirements and the original SLC design requirements. The SLC design requirements are met using one pump based on the minimum flow rate of 26.2 gpm for one-pump operation. Also, Figure 3.4-2 is added to incorporate the minimum solution temperature curve generated by adding 5°F margin instead of 10°F margin to the actual saturation temperature curve as described in the Basis 3.4.3. This is consistent with Final Safety Analysis Report Section 9.3.4.3 and acceptable. The licensee indicated that the Union Pump Co. (UPC) has tested and certified that net positive suction head requirements are satisfied and that vibration readings were well within acceptable levels for two-pump operation (Ref. 7). The associated Basis 3.4 has also been changed to reflect the revised approach and requirements. The staff finds that the licensee's approach meets the requirements of 10 CFR 50.62, paragraph (c)(4), and is, therefore, acceptable.

ATWS/RPT SYSTEM

In Reference 1, the licensee stated that the present RPT system at Duane Arnold will be upgraded to the "Monticello" design which is described in Reference 5. The licensee has installed a combined ARI/RPT redundant trip division and will use redundant breakers at each recirculation pump power feeder. Each ATWS division will trip a separate breaker. Although these breakers are also used for the reactor protection system (RPS) End-of-Cycle (EOC) recirculation pump trip, the EOC breakers installed at the DAEC were purchased with two trip coils. The first trip coil is associated with the RPS EOC/RPT system. The second trip coil is associated with the ATWS/RPT system. There is no interconnecting wiring between the two trip coils.

As stated in Reference 6, the Monticello design is an acceptable reference ATWS/RPT design. The staff concludes that the Duane Arnold ATWS/RPT design is in compliance with the ATWS Rule, 10 CFR 50.62 paragraph (c)(5) and therefore, the associated Technical Specification changes are acceptable.

5.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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7.0 REFERENCES

1. Iowa Electric Light and Power Company letter, R. W. McGaughy to H. Denton, dated February 26, 1987.

2. Iowa Electric Light and Power Company letter, R. W. McGaughy to T. Murley, dated June 1, 1987.

3. Iowa Electric Light and Power Company letter, R. W. McGaughy to T. Murley, dated July 10, 1987.

4. Iowa Electric Light and Power Company letter, W. C. Rothert to T. Murley, dated November 13, 1987.

5. GE Topical Report NEDE-31096-P "Anticipated Transients Without Scram; Response to NRC ATWS Rule 10 CFR 50.62," dated December 1985.

6. Staff SER on GE Topical Report NEDE-31096-P. Letter from Gus Lainas (NRC) to Terry A. Pickens (BWR Owners Group Chairman), dated October 21, 1986.

7. "Duane Arnold Energy Center Standby Liquid Control Dual Pump Test Report," dated April 18, 1985, Union Pump Company, Battle Creek MI.