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SEP 29 1975

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:

The Commission has issued the enclosed Amendment No. 11 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The amendment also incorporates Change No. 12 in the Technical Specifications in accordance with your application dated March 27, 1975. During our review, a few changes were discussed and found mutually acceptable to you and to the NRC staff.

This amendment revises the provisions in the Technical Specifications relating to the temperature limits for the pressure suppression pool water, in accordance with your application for amendment dated March 27, 1975.

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

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 11
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OFFICE	ORB#3	ORB#3	ORB#3	OELD	RL?AD/ORs	ActD:dr
SURNAME	CParrish:kmf	WPaulson	GLear	SA Treby	KRGoller	RSBoyd
DATE	9/9/75	9/9/75	9/11/75	9/24/75	9/25/75	9/29/75

Form AEC-318 (Rev. 9-53) AECM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Iowa Electric Light & Power Company SEP 2 9 1975

cc: w/enclosure

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

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

PROPOSED AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light & Power Company, Central Iowa Power Company, and Corn Belt Power Cooperative (the licensees) dated March 27, 1975, 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.

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

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 12."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Roger S. Boyd, Acting Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 12
Technical Specifications

Date of Issuance: SEP 29 1975

ATTACHMENT TO PROPOSED AMENDMENT NO. 11
CHANGE NO. 12 TO THE TECHNICAL SPECIFICATIONS
FACILITY OPERATING LICENSE NO. DPR-49
DOCKET NO. 50-331

Replace pages 3.7-1 and 3.7-2 with the attached revised pages. Add new pages 3.7-1a and 3.7-48a. The changes are indicated by vertical lines.

LIMITING CONDITIONS FOR OPERATION

3.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained with the following limits.

12 a. Maximum water volume - 61,500 cubic feet

b. Minimum water volume - 58,900 cubic feed

c. Maximum water temperature

(1) During normal power operation-95F.

(2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10F above the normal power operation limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. a. The pressure suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160F or more and the primary coolant pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

2. The primary containment integrity shall be demonstrated as follows:

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110F. Power operation shall not be resumed until the pool temperature is reduced below the normal power
12 operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120F.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor is critical or when the temperature is above 212F and fuel is in the reactor vessel except while performing low

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).	a. Type A Test Primary Reactor Containment Integrated Leakage Rate Test 1) The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of deterioration. In addition, the external surfaces of the torus below the water level shall be inspected on a routine basis for evidence of torus corrosion or leakage. Except for the initial Type A test, all Type A tests shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If a Type A test is completed but the acceptance criteria of Specification 4.7.A.2.a.(9) is not satisfied and repairs are necessary, the Type A test need not be repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool. 12

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO LICENSE NO. DPR-49

(CHANGE NO. 12 TO THE TECHNICAL SPECIFICATIONS)

SUPPRESSION POOL WATER TEMPERATURE LIMITS

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 March 27, 1975, the licensees, Iowa Electric Light and Power Company, Central Iowa Power Company, and Corn Belt Power Coopeartive requested a change in the Technical Specifications appended to Operating License No. DPR-49 for the Duane Arnold Energy Center located near Palo, Iowa. The proposed change in Technical Specifications was submitted in response to our request to the licensee dated February 15, 1975 and is responsive to the guidelines set forth in our letter. We have made additional modifications to these proposed Technical Specifications to improve the clarity and intent of the specification and its basis. The proposed change in Technical Specifications defines new temperature limits for the suppression pool water to provide additional assurance of maintaining primary containment function and integrity in the event of extended relief valve operation.

Discussion

The Duane Arnold Energy Center is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I Containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

1. Steam Vent Clearing Phenomenon

With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 15, 1975 we also requested the licensee to provide information to demonstrate that the torus structure of the primary containment will maintain its integrity throughout the anticipated life of the facility. In its response dated March 27, 1975 the licensee stated that it was investigating this matter and the results of the investigation would be submitted to us on a schedule consistent with the timing which we proposed for licensee response. Because of the apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

2. Steam Quenching Vibration Phenomenon

The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads on the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself, due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event. Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin[1/] exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

Evaluation

The existing Technical Specifications for Duane Arnold limit the torus pool temperature to 95F. This temperature limit assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable opperating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain well below 95F. While this 95F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications.

This action was, as discussed in our February 15 1975 letter, first suggested by the General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on Nobember 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in those communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Implementation of the GE recommended procedures and temperature limits by the proposed change to the Technical Specifications has been evaluated by the NRC staff as follows:

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This new limit and associated requirement to scram the reactor provides additional margin below the 170F temperature related to potential damage to the torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit applicable to surveillance testing of relief valves and RCIC or HPCI operation provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications is a maximum suppression pool water temperature of 120F.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F, the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

d. In addition to the new limits on temperature of the torus pool water, discussion in the Bases includes a summary of operator actions to be taken in the event of a relief valve malfunction. These operating actions are taken in order to avoid the development of temperatures approaching the 170F threshold for potential damage by the steam quenching phenomenon.

Conclusion

We have concluded, based on the consideration 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: SEP 2 9 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COMPANY
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. 11 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Company, 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 revises the provisions in the Technical Specifications relating to the temperature limits for the pressure suppression pool water, in accordance with the licensee's application for amendment dated March 27, 1975.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on July 24, 1975 (40 F.R. 31045). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated March 27, 1975, (2) Amendment No. 11 to License No. DPR-49, with Change No. 12, 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 29th day of September, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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