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DENTON, H. Office of Nuclear Reactor Regulation, Director (post 851125

SUBJECT: Application for amend to License DPR-49 revising Tech Specs to allow two standby liquid control sys boundary valves to be tested w/water instead of air. Fee paid.

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Iowa Electric Light and Power Company

December 3, 1986
NG-86-3525

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Change (RTS-170B)
SLC Boundary Valve Testing
File: A-117, T-23i

Dear Mr. Denton:

We hereby request revision of the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC) in accordance with the Code of Federal Regulations, Title 10, Sections 50.59 and 50.90.

This proposed change (RTS-170B) revises the current Technical Specifications so that the two Standby Liquid Control System (SLCS) boundary valves may be tested with water instead of air. This change was originally part of RTS-170 but held in abeyance pending additional technical justification. The application has been reviewed by the DAEC Operations Committee and DAEC Safety Committee. In accordance with the fee schedule for license amendments (10 CFR 170), a check for $150 is enclosed. The balance of the fee will be paid upon billing.

Pursuant to the requirements of 10 CFR 50.91, a copy of this submittal, including the analysis of no significant hazards considerations, is being forwarded to our appointed state official.

A001 w/check $150.00 #112433
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This application, which consists of three signed originals and 37 copies with their enclosures, is true and accurate to the best of my knowledge and belief.

IOWA ELECTRIC LIGHT AND POWER COMPANY

BY Richard W. McGaughy
Richard W. McGaughy
Manager, Nuclear Division

Subscribed and sworn to Before Me on this 2nd day of December 1986.

Kathleen M. Furman
Notary Public in and for the State of Iowa

RWM/MJM/dmb*

Attachments: 1) Evaluation of Change Pursuant to 10 CFR 50.92
2) Proposed Change RTS-170B including List of Affected Pages

cc: M. Murphy
L. Liu
L. Root
M. Thadani
NRC Resident Office
T. Houvenagle (UD)

EVALUATION OF CHANGE WITH RESPECT TO 10 CFR 50.92

Background:

The criteria of 10 CFR 50 Appendix J, paragraph III.C.2(b). allow for a fluid (water) leak test on piping if it can be shown that the piping will always have a continous water seal. This proposed change to the Duane Arnold Energy Center (DAEC) Technical Specifications will allow the DAEC to perform a leak test on the Standby Liquid Control System (SLCS) primary containment isolation check valves (V-26-8 and V-26-9) using water instead of air.

The two subject valves are in series in the SLCS piping line which runs from the SLCS storage tank down to the reactor vessel. The SLCS injection piping enters the reactor at 60 inches above the vessel zero (i.e., bottom of the vessel) and terminates in the sparger pipe located below the baffle plate, lower shroud and bottom core plate at approximately 170 inches above vessel zero. During the Design Basis Loss of Coolant Accident (instantaneous guillotine severance of a recirculation suction line and LPCI injection valve failure) the reactor core water level falls to approximately 40 inches above vessel zero during the first 30 seconds of the accident. Vessel pressure decreases from 1050 psia to 43 psia in 50 seconds. The pressure in the drywell increases to 43 psia and it contains saturated steam at 270°F. After about three minutes, the water returns to the steady-state level of 300 inches above vessel zero, which is the level of the jet pump intake nozzles. Hence, except for the first three minutes of the accident, the entrance to the SLCS piping will see a continuous water seal. During that three minute period, there are two possible ways that the water in the SLCS piping might be removed: 1) by being boiled off due to the lowering of the saturation temperature as the pressure decreases in the reactor or 2) by being drawn out by the depressurization of the vessel. These two possibilities were evaluated and are addressed below.

(1) Two standard heat transfer calculations were performed to show that the water in the SLCS piping will not boil off. The first calculation was performed for the conditions prior to the accident and shows that the water temperature near the inboard valve (V-26-9) is the same as the drywell atmosphere temperature, approximately 150°F. The piping distance between the reactor vessel and the inboard isolation valve is about eight feet. The piping distance between the inboard isolation valve and the outboard valve (V-26-8) is about 26 feet. Both valves are located outside the reactor pedestal. This calculation has shown that eight feet of piping is long enough such that conduction heating of the water inside the SLCS piping near the inboard valve by the reactor vessel is negligible. Since the outboard valve is even further from the vessel, it also remains at approximately 150°F. This calculation demonstrates that the fluid conditions inside the SLCS piping near the inboard valve are influenced solely by the drywell ambient conditions and are effectively isolated from those inside the reactor vessel.

The second heat transfer calculation was performed for the conditions during the first three minutes of the accident, before reactor vessel reflood occurs. In this case, heating of the water is assumed to occur by exposure of the pipe to the 270°F post-accident drywell temperature.

This calculation shows that the water temperature near the SLCS valves raises less than 10°F. Therefore, based on these two calculations, the water temperature in the two SLCS valves is initially about 150°F and increases to less than 160°F. Since 160°F is much less than the saturation temperature for 43 psia (272°F), boiling does not occur.

(2) The depressurization of the reactor vessel was also studied to determine if the water in the SLCS piping would be evacuated during that three-minute period. A conservative evaluation based on the rate of change of vessel pressure and the geometry of the SLCS piping has concluded that the piping near the inboard valve will remain full of water. The vessel depressurization would, at most, drain only a foot or two of the SLCS piping just outside the reactor vessel.

In summary, our evaluations have shown that water remains in the SLCS piping near the isolation valves at all times, even during the worst case Design Basis Loss-of-Coolant Accident. These conditions permit leaktesting of these valves by a fluid (water) in accordance with 10 CFR Part 50, Appendix J, Paragraph III.C.2.(b).

Iowa Electric Light and Power Company, Docket No. 50-331,

Duane Arnold Energy Center, Linn County, Iowa

Date of Amendment Request: November 26, 1986

Description of Amendment Request: The proposed license amendment would revise the Duane Arnold Energy Center (DAEC) Technical Specification Table 3.7-2 of Section 3.7 to show that the leak test on Standby Liquid Control boundary valves (V-26-8 and V-26-9) will be performed with water instead of air.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards (10 CFR 50.92(c)) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

In reviewing this proposed request for Technical Specification change we have concluded that this amendment:

(1) The probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously analyzed in the FSAR is not increased. Performing this leaktightness test using water instead of air as the test medium will, by the above evaluations, reliably demonstrate that primary containment integrity is maintained post-accident. In addition, the probability or consequences of any accident previously analyzed is not affected by changing the test medium for the SLCS isolation valves.

(2) does not create the possibility of a new or different kind of accident because this proposal does not involve any changes in hardware at the plant or in the operation of existing equipment. Switching the testing fluid from air to water, the actual operating fluid, cannot create the possibility of a new type of acccident.

(3) does not involve a significant reduction in a margin of safety because testing these valves with water instead of air provides equal assurance that a significant leak is not present. The margin of safety in the reliability that a significant leak does not exist is maintained. The margin of safety in the reliability that these valves will perform their safety function is not affected.

Therefore, this proposed license amendment is judged to involve no significant hazards consideration.

Local Public Document Room Location: Cedar Rapids Public Library, 500 First Street SE, Cedar Rapids, Iowa 52401

Attorney for Licensee: Jack Newman, Kathleen H. Shea, Newman and Holtzinger, 1615 L Street NW, Washington, DC 20036

PROPOSED CHANGE RTS-170B TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached, new pages.

Summary

The following list of proposed changes is in the order that the changes appear in the Technical Specifications. The List of Affected Pages is presented following this list of changes.

This proposed change to the Technical Specifications will allow the DAEC to perform a leak tightness test on the Standby Liquid Control System (SLCS) primary containment isolation check valves (V-26-8 and V-26-9) using water instead of air. This change was originally part of RTS-170 but was held in abeyance pending additional technical justification. A leak test, using water as the test medium, is allowed by 10 CFR 50 Appendix J, Paragraph III.C.2.(b) if it can be shown that the piping will have a continuous water seal. This requirement is met as outlined in the following paragraphs.

The two subject valves are in series in the SLCS piping line which runs from the SLCS storage tank down to the reactor vessel. The SLCS injection piping enters the reactor at 60 inches above the bottom of the vessel (i.e., vessel zero) and terminates in the sparger pipe located below the baffle plate, lower shroud and bottom core plate at approximately 170 inches above vessel zero. During the Design Basis Loss of Coolant Accident (instantaneous guillotine severance of a recirculation suction line and LPCI injection valve failure) the reactor core water level falls to approximately 40 inches above vessel zero during the first 30 seconds of the accident. Vessel pressure decreases from 1050 psia to 43 psia in 50 seconds. The pressure in the drywell increases to 43 psia and it contains saturated steam at 270°F. After about three minutes, the water returns to the steady-state level of 300 inches above vessel zero, which is the level of the jet pump intake nozzles. Hence, except for the first three minutes of the accident, the entrance to the SLCS piping will see a continuous water seal. During that three minute period, there are two possible ways that the water in the SLCS piping might be removed: 1) by being boiled off due to the lowering of the saturation temperature as the pressure decreases in the reactor or 2) by being drawn out by the depressurization of the vessel. These two possibilities were evaluated and are addressed below.

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In summary, our evaluations have shown that water remains in the SLCS piping near the isolation valves at all times, even during the worst case Design Basis Loss-of-Coolant Accident. These conditions permit leaktesting of these valves by a fluid (water) in accordance with 10 CFR Part 50, Appendix J, Paragraph III.C.2.(b).

This change will require the modification of only one page in the Technical Specifications, page 3.7-23. A superscript 1 is added to penetration #42 in Table 3.7-2 to refer to Note 1 on page 3.7-24 for instructions to test with water.

LIST OF AFFECTED PAGE

3.7-23