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DESCRIPTION: Ltr notarized 12-15-75 raequests for amdt to OL/DPR -49/Tech Specs for Appendices A & B to Lic.& trans the following:

ENCLOSURES: Proposed changes in Tech Specs (Appendices A & B) for Duane Arnold Energy Center....

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IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office

CEDAR RAPIDS, IOWA

December 15, 1975
IE-75-1357

LEE LIU
VICE PRESIDENT – ENGINEERING

50-331

Mr. B. C. Rusche, Director
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Rusche:

Transmitted herewith, in accordance with the requirements of 10CFR50.59 and 50.90, is an application for amendment of DPR-49 to incorporate proposed changes in the Technical Specifications (Appendices A and B to License) for the Duane Arnold Energy Center (DAEC), described in the enclosures hereto.

This proposed change has been reviewed and approved by the DAEC Operations Committee and the DAEC Safety Committee and 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
Encls.
cc: W/encls.
D. Arnold
J. Keppler
J. Newman
W. Paulson
R. Bevan

Sworn and Subscribed to before me on this 15th day of December, 1975.

Georgia F. Marlowe
Notary Public in and for the State of Iowa.

Georgia F. Marlowe
NOTARY PUBLIC
State of Iowa
Commission Expires
September 30, 1976

14312

PROPOSED CHANGE ETS-16 TO DAEC TECHNICAL SPECIFICATIONS

Received W/Ltr Dated 12-15-75

I. Affected Technical Specifications

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

Specification 4.1.1.6.B

"Location: Artificial substrates will be installed at Site 2, above the plant intake, and at Site 3, below the plant, and in the discharge canal."

II. Proposed Change in Technical Specifications

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

Delete "and in the discharge canal".

III. Justification for Proposed Change

In order for periphyton growth to take place the substrates require continual submergence in water for a period of from two to four weeks. The DAEC mode of operation is such that there is not always water in the discharge canal since blowdown from the cooling towers is secured while condenser chlorination is taking place. The original purpose for placing the substrates in the discharge canal was to determine what affect chlorine had on periphyton growth prior to the time the discharge was diluted with river water. The intent of the Technical Specifications will still be met since substrates are installed above the plant intake and below the plant discharge to monitor any affect plant discharge has on the river.

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.

4.0 ENVIRONMENTAL SURVEILLANCE AND SPECIAL STUDIES

4.1.1 Specification (Cont'd.)

4.1.1.3 Plankton Studies

A. Frequency: Twice per month routinely and as necessary when conditions warrant.

B. Location: At all four river locations and the discharge canal.

C. Analyses to be made: Numbers and kinds (to genus whenever possible) of organisms present.

4.1.1.4 Bacteriological Studies

A. Frequency: Twice per month. Additional determinations of fecal coliforms will be conducted on samples from the effluent from the station's sewage treatment plant.

B. Location: At all four river locations and the discharge canals.

C. Analyses to be made:

1. Total plate count (20 C.)
2. Total coliform (MF)
3. Fecal coliform (MF)
4. Fecal streptococci (MF)

4.1.1.5 Benthic (bottom organism) Studies

A. Frequency: Quarterly

B. Location: At all four river sites

C. Analysis: Kinds (to genus whenever possible) and numbers of organisms present will be determined. Sediment type will also be determined.

4.1.1.6 Periphyton

A. Frequency: Three times per year during spring, summer and fall, as available.

B. Location: Artificial substrates will be installed at Site 2, above the plant intake, and at Site 3, below the plant.

C. Analyses to be made: Substrates will be removed after two weeks to one month. The biomass and generic composition will be determined.

PROPOSED CHANGE ETS-17 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

Appendix B of the Technical Specifications for the DAEC (DPR-49) provide as follows:

Table 4.3-1, "Environmental Radioactivity Monitoring Program for the Duane Arnold Energy Center", provides, among other sample points, Sample Points 66 through 68 as "farms within 10 miles of the site" used for obtaining soil, vegetation and milk samples.

II. Proposed Change in Technical Specifications

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

> Delete Sample Points "67" and "68" and add Sample Point "95".

The sample points for soil (p. 4.3-8) will now be as follows: "15, 16, 62-64, 66, 71-73, 93-95 and 74".

The sample points for vegetation (p. 4.3-8) will now be as follows: "62-64, 66, 71-73 and 93-95".

The sample points for milk (pp. 4.3-10 and 4.3-11) will now be as follows: "62, 63-64, 66, 71-72, 73, 93, 94 and 95".

The above changes were also made on Figure 4.3-1, "Radiological Environmental Monitoring Program Sampling Stations". In addition, Sampling Point 70 was deleted from Figure 4.3-1.

This deletion (Sampling Point 70) had been previously approved by the NRC for Table 4.3-1 (Amendment No. 7, Change 8 to Operating License No. DPR-49, dated May 16, 1975), but was inadvertently not deleted from the figure at that time.

III. Justification for Proposed Change

The farms identified as Sample Points 67 and 68 no longer have milk available on a reliable enough schedule to be considered for the Environmental Radioactivity Monitoring Program. One new source identified as Sample Point 95 has been found and included in the monitoring program. Since the soil and vegetation monitoring programs use the same sample points as the milk program, in the interest of standardization, the sample point changes have been made there also. For these reasons, the above changes are proposed.

IV. Review Procedures

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 4.3-1 (Continued)

ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM FOR THE DUANE ARNOLD ENERGY CENTER

SAMPLING DESCRIPTION			SAMPLE FREQUENCY		ANALYSIS	REMARKS
Type of Sample	Sample Point	Sampling Point Description	Preoperational Program	Operational Program		
Bottom Sediments	49 50 51 61	Lewis Access Plant intake Plant Discharge One-half mile below plant discharge	Monthly	Semi-Annually	Gross alpha Gross beta $-K^{40}$ Gamma isotopic analysis ^{90}Sr	Routine gross alpha and gross beta $-K^{40}$ during preoperational phase.
Soil	15 16 62-64 66 71-73 93-95 74	On-site On-site Farms (within 10 miles of the site) that raise food crops Irrigated farm downstream of plant	Quarterly	Annual during growing season	Gross alpha Gross beta $-K^{40}$ Gamma isotopic analysis ^{90}Sr	Routine gross alpha and gross beta $-K^{40}$ during preoperational phase. Surface sample from undisturbed area.
Vegetation	62-64 66 71-73 93-95	Farms that raise food crops	Annually at harvest time (as available)	Annually at harvest time	Gross alpha Gross beta $-K^{40}$ Gamma isotopic analysis ^{90}Sr	Routine gross alpha and gross beta $-K^{40}$ during preoperational phase. Only the edible portion of crops will be analyzed.
Meat and Poultry		Farms (within 10 miles of the site) that raise poultry or animals for human consumption	As Available	Annually during or immediately following grazing season	Gamma isotopic analysis on edible portions	The specific location of these samples will vary with availability

TABLE 4.3-1 (Continued)

ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM FOR THE DUANE ARNOLD ENERGY CENTER

SAMPLING DESCRIPTION			SAMPLE FREQUENCY		ANALYSIS	REMARKS
Type of Sample	Sample Point	Sampling Point Description	Preoperational Program	Operational Program		
Milk	62	Control Farm near Brendon, Iowa	Monthly	Weekly	^{131}I	Preoperationally ^{131}I will be analyzed routinely on a monthly basis and more frequently if ^{131}I is detected or suspected. Operationally during the grazing season samples from locations 63, 94 and 93 will be analyzed individually. Operationally during the grazing season samples from locations 64, 66, 67, 68, 71 & 72 will be composited and analyzed. If the composite sample is greater than 2.4 pCi/l the location will be resampled and samples analyzed individually. Operationally during the grazing season samples from locations 62 and 73 will be composited and analyzed. If the composite sample is greater than 2.4 pCi/l the location will be resampled and samples analyzed individually.
	63-64	Dairy farms within 10 mi. of site				
	66	Dairy farm within 10 mi. of site				
	71-72	Dairy farms within 10 mi. of site				
	73	Control farm near Amana, Iowa				
	94	Dairy farm within 10 mi. of site				
	93	Dairy farm within 10 mi. of site				
	95	Dairy farm within 10 mi. of site				

TABLE 4.3-1 (Continued)

ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAM FOR THE DUANE ARNOLD ENERGY CENTER

SAMPLING DESCRIPTION			SAMPLE FREQUENCY		ANALYSIS	REMARKS
Type of Sample	Sample Point	Sampling Point Description	Preoperational Program	Operational Program		
Milk	62	Control farm near Brendon, Iowa	Monthly	Monthly	^{89}Sr	Operationally during the grazing season a portion of the weekly sample from each location will be composited for analysis.
	63-64	Dairy farms within 10 mi. of site			^{90}Sr ^{137}Cs	
	66	Dairy farms within 10 mi. of site			$^{140}Ba - ^{140}La$	
	71-72	Dairy farms within 10 mi. of site			Elemental Ca	
	73	Control farm near Amana, Iowa		Monthly	^{131}I	Operationally during the non-grazing season a sample from all lotions except locations 62 and 73 will be composited and analyzed.
	94	Dairy farm within 10 mi. of site				
	93	Dairy farm within 10 mi. of site				Operationally during the non-grazing season a sample from locations 62 & 73 will be composited and analyzed.
	95	Dairy farm within 10 mi. of site				

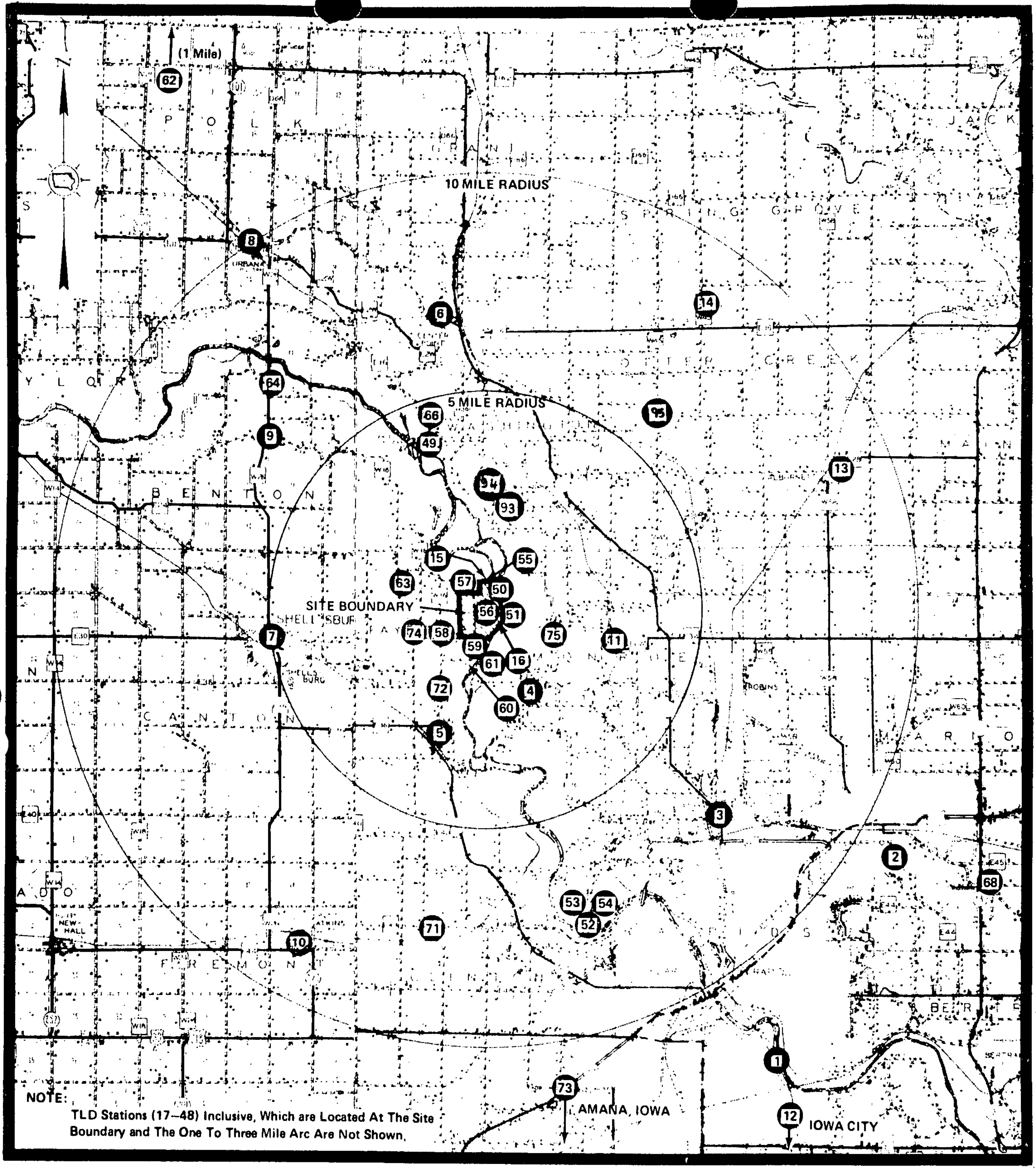

FIGURE 4.3-1 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SAMPLING STATION

PROPOSED CHANGE RTS-35 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

The Technical Specifications for the DAEC (DPR-49, Appendix A) do not provide for periodic inspection of RPV Seismic Stabilizer assemblies and attachments.

II. Proposed Change in Technical Specifications

The licensees of DPR-49 propose the following additions to Specification 4.6.G, Surveillance Requirements, in the Technical Specifications covering the subject set forth in I, above:

"7. During each plant refueling outage all eight RPV Seismic Stabilizer assemblies and attachments will be inspected as follows:

a. Visually inspect stabilizer assembly parts for deformation and cracking.

b. Verify that all clevis pin retainers are in place.

c. Verify that all drawbar set screws are in place.

d. Visually inspect stabilizer gusset plate welds.

e. Visually inspect stabilizer support-to-RPV welds such that all four welds are inspected during the regular 10-year in-service inspection interval."

III. Justification for Proposed Change

This change is being submitted in compliance with a commitment made by Iowa Electric Light and Power Company to the Nuclear Regulatory Commission regarding periodic examination of the RPV Seismic Stabilizers. (Letter IE-74-2002, C. Sandford, Executive Vice President, Iowa Electric Light and Power Company to J. Keppler, Regional Director, Region III, U. S. Atomic Energy Commission, dated August 14, 1974.)

IV. Review Procedures

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.

remaining system components or piping in this category shall be examined to the extent practical as specified in that examination category.

6. Detailed records of each inspection including the preoperational base line inspection, shall be maintained to allow comparison and evaluation or future inspection. The records shall conform to the requirements of IS-600 of Section XI of the ASME Boiler and Pressure Vessel Code.

7. During each plant refueling outage all eight RPV Seismic Stabilizer assemblies and attachments will be inspected as follows:

a. Visually inspect stabilizer assembly parts for deformation and cracking.

b. Verify that all clevis pin retainers are in place.

c. Verify that all drawbar set screws are in place.

d. Visually inspect stabilizer gusset plate welds.

e. Visually inspect stabilizer support-to-RPV welds such that all four welds are inspected during the regular 10-year in-service inspection interval.

PROPOSED CHANGE RTS-36 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Specification 6.1-1, p. 6.1-1

"The Chief Engineer has primary responsibility for the safe operation of the DAEC-1 plant, and reports, under the Executive Vice President, to the General Production Manager."

Specification 6.5.1.4.g, p. 6.5-3

"Investigate reported or suspected violations of Technical Specifications and Operating Procedures. These investigations will include reporting, evaluation and recommendations to prevent recurrence to the Chief Engineer, the General Production Manager and the Chairman of the DAEC Safety Committee."

Specification 6.5.2.2, p. 6.5-5

"Membership in the Safety Committee shall be by appointment by the Executive Vice President, and shall consist of eight (8) persons, two of whom shall be designated as Chairman and Vice Chairman, respectively."

Specification 6.6.1, p. 6.6-1

"Any abnormal occurrence shall be reported immediately to the Chief Engineer and to the General Production Manager, and promptly reviewed by the Operations Committee."

Specification 6.6.3, p. 6.6-1

"Copies of all such reports shall be submitted to the Safety Committee for review and to the General Production Manager for review and approval of any recommendations."

Specification 6.7.2, p. 6.7-1

"An immediate report shall be made to the General Production Manager and the Safety Committee. The General Production Manager shall promptly report the circumstances to the AEC as specified in Subsection 6.12, Plant Reporting Requirements."

Specification 6.7.3, p. 6.7-1

"A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Operations Committee. This report shall be submitted to the General Production Manager and the Safety Committee. Appropriate analyses or reports will be submitted to the AEC by the General Production Manager as specified in Subsection 6.12, Plant Reporting Requirements."

II. Proposed Changes in Technical Specifications

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

Change the title "General Production Manager" to "Vice President-Generation" as indicated by the underlined portions.

In Specifications 6.1-1 and 6.5.2.2 change "Executive Vice President" to "Chairman of the Board and President".

III. Justification for Proposed Changes

The new corporate position of Vice President-Generation was established by the Iowa Electric Light and Power Company Board of Directors on February 4, 1975. The Vice President-Generation is responsible for all the activities previously assigned to the Production Department pertaining to the operation, maintenance and facility expansion activities of the electric generating properties.

The corporate position of Executive Vice President no longer exists at Iowa Electric Light and Power Company. Those responsibilities described in the Technical Specifications which were previously those of the Executive Vice President have now been assumed by the Chairman of the Board and President.

For these reasons these changes are proposed.

IV. Review Procedures

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.

6.0 ADMINISTRATIVE CONTROLS

6.1 MANAGEMENT - AUTHORITY AND RESPONSIBILITY

6.1.1 The Chief Engineer has primary responsibility for the safe operation of the DAEC-1 plant, and reports, under the Chairman of the Board and President, to the Vice President-Generation.

in procedures in b. above or may constitute an unreviewed safety question.

f. Review plant operations to detect any potential safety hazards.

g. Investigate reported or suspected violations of Technical Specifications and Operating Procedures. These investigations will include reporting, evaluation and recommendations to prevent recurrence to the Chief Engineer, the Vice President-Generation and the Chairman of the DAEC Safety Committee.

h. Submit proposed procedure changes having safety significance and unreviewed safety questions resulting from a. through g. above to the DAEC Safety Committee for review.

i. Perform special reviews and investigations and rendor reports thereon as requested by the Chairman of the Safety Committee.

2. Membership

Membership in the Safety Committee shall be by appointment by the Chairman of the Board and President, and shall consist of eight (8) persons, two of whom shall be designated as Chairman and Vice-Chairman, respectively. Not more than a minority of a quorum may have concurrent on-site line responsibility for the operation of the DAEC and no such member shall be elegible to be Chairman or Vice Chairman.

3. Qualifications of Membership

Members of the Safety Committee shall collectively have or have access to applicable technical and experience expertise in the following areas:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and Radiochemistry
d. Instrumentation and Control
e. Radiation Protection
f. Mechanical and Electrical Engineering
g. Nuclear Safety

6.6 REPORTABLE OCCURRENCE ACTION 13

6.6.1 Any reportable occurrence shall be reported immediately 13
to the Chief Engineer and to the Vice President-Generation, and promptly reviewed by the Operations Committee.

6.6.2 The Operations Committee shall prepare a separate report for each reportable occurrence. This report shall include 13
an evaluation of the cause of the occurrence, a record of the corrective action taken, and also recommendations for appropriate action to prevent or reduce the probability of a recurrence.

6.6.3 Copies of all such reports shall be submitted to the Safety Committee for review and to the Vice President-Generation for review and approval of any recommendations.

6.7 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.7.1 If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed when authorized by the AEC.

6.7.2 An immediate report shall be made to the Vice President-Generation and the Safety Committee. The Vice President-Generation shall promptly report the circumstances to the AEC as specified in Subsection 6.12, Plant Reporting Requirements.

6.7.3 A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence shall be prepared by the Operations Committee. This report shall be submitted to the Vice President-Generation and the Safety Committee. Appropriate analyses or reports will be submitted to the AEC by the Vice President-Generation as specified in Subsection 6.12, Plant Reporting Requirements.

PROPOSED CHANGE RTS-45 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Specification 3.8.C.3, p. 3.8-6

"If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours."

II. Proposed Change in Technical Specifications

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

Change "3.5.C" to "3.8.C".

III. Justification for Proposed Change

Typing error.

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.

LIMITING CONDITIONS FOR OPERATION

C. Emergency Service Water System

1. Except as specified in 3.8.C.2 below, both emergency service water system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

11

2. From and after the date that one of the emergency service water system pumps or loops is made or found to be inoperable for any reason, reactor operation must be limited to seven days unless operability of that system is restored within this period. During such seven days all active components of the other Emergency Service Water System shall be operable, provided the requirements of 3.5.G are met.

3. If the requirements of 3.8.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

C. Emergency Service Water System

1. Emergency Service Water Subsystem Testing

a. Simulated automatic actuation test.	each refueling outage
b. Pump and motor operated valve operability	once/3 months
c. Flow Rate Test	
Each emergency service water pump shall deliver at least that flow determined from Figure 4.8.C-1 for the existing river water temperature.	after major pump maintenance and every month, except weekly during periods of time the river water temperature exceeds 80°F.

2. When one emergency service water system pump or loop becomes inoperable, the operable pump and loop and diesel-generator required for operation of such components shall be demonstrated to be operable immediately and daily thereafter.

PROPOSED CHANGE RTS-46 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Specification 2.2 Bases, pp. 1.2-5 and 1.2-6

> "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 a maximum vessel pressure of 1295 psig (at vessel bottom) if a pressure scram is assumed."

Specification 3.6.D and 4.6.D Bases, pp. 3.6-23 and 3.6-24

> "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 a maximum vessel pressure of 1295 psig (at vessel bottom) if a pressure scram is assumed."

II. Proposed Changes in Technical Specifications

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

> In both of the sentences above, change "pressure scram" to "flux scram" and "1295 psig" to "1292 psig".

III. Justification for Proposed Changes

Pressure scram was inadvertently used rather than flux scram when the Technical Specifications were being developed. Section 4.2.3 of the Safety Analysis with Bypass Holes Plugged submitted to the NRC on June 10, 1975 shows that neglecting the direct scram results in a maximum pressure of 1292 psig. For this reason, the changes described in part II are proposed.

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.

valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1292 psig (at vessel bottom) if a flux scram is assumed. This results in 80 psig to the code allowable overpressure limit of 1375 psig. In addition, the same event was analyzed to determine the number of installed valves which must open to limit peak pressure to 1350 psig (25 psig margin). The results of this analysis show that four valves must open if a neutron flux scram is assumed.

To meet the second design basis, the total safety/relief capacity has been divided into 6 relief valves and 2 safety valves. The analysis of the plant isolation transient (turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in FSAR paragraph 14.5.1.2 and Figure 14.5-3. This analysis shows that the 6 relief valves limit pressure at the safety valves to 1196 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1234 psig which is 141 psig below the allowed vessel overpressure of 1375 psig.

the direct scram (valve position scram) results in a maximum vessel pressure of 1292 psig (at vessel bottom) if a flux scram is assumed. This results in 80 psig margin to the code allowable overpressure limit of 1375 psig. In addition, the same event was analyzed to determine the number of installed valves which must open to limit peak pressure to 1350 psig (25 psig margin). The results of this analysis show that five valves must open if a neutron flux scram is assumed or six valves must open if a pressure scram is assumed.

To meet the power generation design basis, the total safety/relief capacity has been divided as described previously in Subsection 2.2. 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. This analysis shows that the 6 relief valves limit pressure at the safety valves to 1196 psig, well below the setting of the safety valves. Therefore, the safety valves will not open. This analysis shows that peak system pressure is limited to 1234 psig which is 141 psig below the allowed vessel overpressure of 1375 psig.

Experience in relief and safety valve operation shows that a testing of 50 percent of the valves per year is adequate to

"The requirements of ANSI N18.7-1972, Section 4.3.1 are fulfilled by periodic Safety Committee audits of design change safety evaluations completed under the provisions of Paragraph 50.59(b), Part 50, Title 10, Code of Federal Regulations."

(B) Change the heading of Specification 6.5.2.7 from "Meetings, Minutes and Records" to "Meeting Records".

Change Specification 6.5.2.7 to read "Minutes of all meetings of the Safety Committee shall be prepared and retained."

(C) Add to Specification 6.5.2.8 the following item:

"e. Design Change Request Safety Evaluation."

III. Justification for Proposed Changes

(A) and (C) The Charter requirements were initially established under the assumption that design changes would be initiated within the DAEC operating organization and without Safety Committee review - would not have received an independent review. Corporate policy established the requirements that all design changes be developed and approved within the Engineering organization which does not have direct responsibility for operation of the plant. This policy is applied to all components and systems of the plant whether they are part of or auxiliaries of the Nuclear Steam Supply System or not. The Engineering organization is responsible for the Safety Evaluation in conjunction with the design specifications and related design and licensing document changes. The complete design change package which includes the aforementioned documents is reviewed and approved by the Project Engineer, the Manager of Mechanical/Nuclear Engineering, and the Manager, Quality Assurance, or their designated alternates. The approved design change is then forwarded to the on-site Operations Committee for their review and approval. Design changes that would possibly involve an unreviewed safety question are forwarded to the Safety Committee. It is considered that the above action by off-site personnel results in an independent review process intended by ANSI 18.7, Section 4.3.1. Periodic audits by the Safety Committee of design change evaluations is consistent with the intent of ANSI 18.7 when coupled with the independent review conducted by the Engineering Department.

(B) Meetings of the Safety Committee are generally called on a time schedule such that members are notified by verbal communication. Copies of the meeting minutes state what subjects were covered at the meeting so the written notice does

PROPOSED CHANGE RTS-56 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

(A) Specification 6.5.2.1, pp. 6.5-4

"The Charter shall incorporate specific provisions which meet the requirements of an IRAG as specified in ANSI N18.7-1972, Sections 4.1-4.4, the specifications of this Subsection 6.5.2 and such other provisions as may be necessary, including provision for its amendment from time to time, to assure that it governs an orderly and effective review and audit process throughout the service life of the plant."

(B) Specification 6.5.2.7, Meetings, Minutes and Records, pp. 6.5-7

"Notices to members and minutes of all meetings of the Safety Committee shall be prepared and retained along with copies of all documentary and supporting materials referenced. (Copies of referenced materials which are publicly available need not be retained with meeting records.)"

(C) Specification 6.5.2.8, pp. 6.5-7 and 6.5-8

"Subjects to be audited shall include:

a. Surveillance status of DAEC systems and equipment as they relate to safety.

b. Operational status of DAEC systems and equipment as they relate to safety.

c. Personnel training.

d. Radiological and environmental effects of plant operation."

II. Proposed Changes in Technical Specifications

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

(A) Add to Specification 6.5.2.1, the following:

not serve a useful purpose. All referenced material is retained in appropriate files and is not necessary to be part of the meeting minutes.

IV. Review Procedures

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.

6.5.2 Safety Committee

1. Charter

Iowa Electric Light and Power Company management shall cause to be written, and shall approve, a charter for constituting a management Safety Committee which shall have responsibility and authority for review and audit of DAEC plant operations to verify that operation of the plant is consistent with company policy and rules, approved operating procedures and license provisions, to review proposed plant changes, tests, and procedures, to verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events.

The charter shall incorporate specific provisions which meet the requirements of an IRAG as specified in ANSI N18.7-1972, Sections 4.1 - 4.4, the specifications of this Subsection 6.5.2 and such other provisions as may be necessary, including provisions for its amendment from time to time, to assure that it governs an orderly and effective review and audit process throughout the service life of the plant. The requirements of ANSI N18.7-1972, Section 4.3.1, are fulfilled by periodic Safety Committee audits of design change safety evaluations completed under the provisions of Paragraph 50.59(b), Part 50, Title 10, Code of Federal Regulations.

7. Meeting Records

Minutes of all meetings of the Safety Committee shall be prepared and retained.

8. Subjects Requiring Audit by Safety Committee

Subjects to be audited shall include:

a. Surveillance status of DAEC systems and equipment as they relate to safety.

b. Operational status of DAEC systems and equipment as they relate to safety.

c. Personnel training.

d. Radiological and environmental effects of plant operation.

e. Design Change Request Safety Evaluation.

9. Authority of Safety Committee

The charter shall specify the extent of authority carried by the Committee's actions and conclusions.

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PROPOSED CHANGE RTS-57 TO DAEC TECHNICAL SPECIFICATIONS

I. Affected Technical Specifications

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

Table 3.2-A (Instrumentation that initiates Primary Containment Isolation) does not state what the Minimum Number of Operable Instrument Channels Per Trip System should be for Reactor Low Pressure (Shutdown Cooling Isolation).

II. Proposed Change in Technical Specifications

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

The Minimum Number of Operable Instrument Channels Per Trip System should be "1".

III. Justification for Proposed Change

Draft copies of the DAEC Technical Specifications indicated that one operable instrument channel per trip system was the minimum number allowable. The number was evidently inadvertently deleted when the final copy was prepared.

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 3.2-A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided by Design	Action(2)
2 (6)	Reactor Low Water Level	≥ +12" Indicated Level (3)	4 Inst. Channels	A
1	Reactor Low Pressure (Shutdown Cooling Isolation)	≤ 135 psig	2 Inst. Channels	C
2	Reactor Low-Low-Water Level	At or above -38.5" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤ 2.0 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 X Normal Rated Power Background (8)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 880 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel/Turbine Bldg. High Temperature	≤ 200 deg. F	4 Inst. Channels	B
1	Reactor Cleanup System High Diff. Flow	≤ 40 gpmd	2 Inst. Channel	D
1	Reactor Cleanup System High-High Temperature	≤ 140°F	1 Inst. Channel	D