9 FEB 1987

Docket No. 50-331

MEMORANDUM FOR: Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

FROM: Robert A. Gilbert, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

SUBJECT: MEETING WITH IOWA ELECTRIC LIGHT AND POWER

SEE REPORT.

NRC staff personnel and contractors met with representatives of Iowa Electric Light and Power (IELP) in Bethesda, Maryland on January 20-21, 1987. The purpose of the meeting was to discuss IELP's proposed responses to staff questions transmitted to them on December 2, 1986 relating to their proposed 2nd 10-year IST Program. Participants in the meeting are shown in Enclosure 1.

IELP's responses are furnished as Enclosure 2. As each response was discussed, notes were taken by the staff's contractor documenting the agreements reached. These notes will be complied as meeting minutes and formally transmitted to IELP in the near future. IELP will respond to some further staff questions which arose as a result of the discussions and, based on the agreements needed, will transmit Revision 8 of their IST Relief Request within 60 days of the receipt of the meeting minutes.

Original signed by

Robert A. Gilbert, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

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50-331

MEETING WITH IOWA
ELECTRIC LIGHT AND POWER

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Enclosure 1

Participants at Meeting on January 20-21, 1987

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INTRODUCTION

In their letter dated December 2, 1986, the NRC staff requested additional information pertinent to their review of Revision 7 to the Inservice Testing (IST) Program for Pumps and Valves dated December 31, 1985. The following information provides responses to the NRC questions.

A. GENERAL QUESTIONS AND COMMENTS

NRC Question No. 1

Relief Requests that reference the FSAR, Technical Specifications, and other documents should be expanded to provide a brief discussion of the applicable technical information contained in the applicable document.

Iowa Electric Response

The IST Program does reference specific portions of the FSAR, Technical Specifications and other documents. Your specific needs will be discussed during the upcoming meeting of January 20 and 21.

NRC Question No. 2

Are all valves that are Appendix J, Type C, leak tested included in the IST Program as Category A or A/C? Those valves performing both a pressure boundary isolation function and a containment isolation function must be leak tested to both the Section XI and Appendix J requirements.

Iowa Electric Response

Yes.

NRC Question No. 3

Have all valves that have a required fail-safe position been included in the IST Program?

Iowa Electric Response

No, in accordance with Article IWV-1000, only valves required to mitigate the consequences of an accident or required for safe shutdown of the reactor are included in the Program. However, there are some valves which do not meet the IWV-1000, ASME Section XI criteria, yet do have fail-safe positions.

B. NEUTRON MONITORING

NRC Question No. 1

Is the maximum stroke time given for valves TIP-BAL A, B, and C a typographical error? Page 1 of the IST Program does not agree with Relief Request VR-34.

Iowa Electric Response

No, the maximum stroke time given for valves TIP-BAL A, B, and C are not typographical errors, nor are the maximum stroke times in conflict with Relief Request VR-34. Although the maximum stroke time is 5 seconds, the reason they are not in conflict is that the valves normally stroke in less than 2 seconds. Relief Request VR-34 changes the requirement for increasing test frequency of valves from that which is delineated in Subparagraph IWV-3417(a) to "if a measured stroke time increase of a 100% or more from a previous test is observed and the stroke time is greater than 2 seconds." The maximum allowable stroke time for the valves are specified by the owner, as required in Subparagraph IWV-3413(a) and are not established by Relief Request VR-34.

C. CONDENSATE AND DEMINERALIZED WATER

NRC Question No. 1

Are Category A valves V-09-065 and -111 passive valves? If so, relief from exercising is unnecessary according to IWV-3700.

Iowa Electric Response

Yes, these valves are passive and will be removed from Relief Request VR-36. Since valve V-30-287 is also passive, Relief Request VR-36 will be withdrawn.

D. REACTOR BUILDING COOLING WATER

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valves MO-4841A and B quarterly during power operation.

Iowa Electric Response

These valves are primary containment isolation valves for the reactor building closed cooling water system. During power operation, the RBCCW system supplies cooling water to components inside the drywell, including the reactor recirculation pumps and motors. Exercising the subject valves would interrupt cooling water flow to the reactor recirculating pump and motor heat exchangers. These valves will not be exercised during normal operation because interruption of flow would cause damage to the reactor recirculation pumps and motors.

E. RHR AND EMERGENCY SERVICE WATER

NRC Question No. 1

Why is a maximum stroke time identified for valves CV-1956A and B and then relief requested from stroke timing?

Iowa Electric Response

Since the valves cannot be timed due to lack of position indication, no maximum stroke time should be identified. The stroke times presently in the Program will be removed.

NRC Question No. 2

Review the safety function of the following valves to determine if they should be included in the IST Program and categorized as indicated.

Category B	Category C
MO-1947 (C-6)	V-13-4 (F-7)
MO-2046 (C-5)	V-13-15 (G-7)
MO-1998A (B-7)	
MO-1998B (B-7)	

Iowa Electric Response

MO-1947 - We will add to the IST Program as a Category B valve. A relief request will be submitted in the near future.

MO-2046 - We will add to the IST Program as a Category B valve. A relief request will be submitted in the near future.

MO-1998A - Does not perform any safety function. If the valve is closed, then discharge will flow to radwaste dilution. PSE-2079A and PSE-2079B have had internals removed.

MO-1998B - See MO-1998A above.

V-13-04 - The line which contains this valve does not perform a safety function except as a Class 3 pressure boundary. The motor operated valve downstream is the Class 3 to non-class boundary (See UFSAR Section 5.4.7.1).

V-13-15 - See V-13-04 above.

F. NUCLEAR BOILER

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valves CV-4428 and CV-4429 quarterly during power operation. What is the safety-related function of these valves?

Iowa Electric Response

The valves are used to vent the reactor vessel head during startup and shutdown. The safety-function of the valves is to close. Exercising one of these valves during normal operation leaves the other valve as the only barrier between the reactor vessel and the drywell sump. Any leakage through the closed valve could potentially pressurize the drywell. In addition, operating procedures prohibit operation of these valves during power operation.

NRC Question No. 2

Provide a detailed technical justification for not full-stroke exercising valves MO-4441 and MO-4442 quarterly during power operation.

Iowa Electric Response

Feedwater valves MO-4441 and MO-4442 cannot be exercised during reactor operation because the feedwater system is needed to maintain primary coolant inventory. Also, if a feedwater isolation valve was closed during operation, the feedwater nozzle and spargers would undergo a severe thermal shock when feedwater was restored. This thermal shock could cause cracking and possible failure of the spargers and nozzles.

NRC Question No. 3

Can valves PSV-4439A, B, C, D, E, and F be exercised during cold shutdown? Are these valves simple check valves?

Iowa Electric Response

No, during a relief valve discharge, these valves must be closed. After a relief valve discharge, the steam remaining in the relief valve discharge line will condense and try to draw a vacuum in the discharge line. These relief valves (vacuum breakers) open to the discharge line thus relieving the vacuum condition. These valves have no external means of actuation for exercising. The only practical method of exercising these valves open and closed is by manually pushing the disk from its seat. Since this requires access to the valves, which are located in the drywell, these valves will be exercised concurrent with the setpoint verification tests, in accordance with Subarticle IWV-3510 to ASME Section XI.

NRC Question No. 4

Provide a detailed technical justification for not full-stroke exercising valves V-14-001 and V-14-003 quarterly during power operation.

Iowa Electric Response

V-14-001 and V-14-003 cannot be exercised during reactor operation because the feedwater system is needed to maintain primary coolant inventory. Also, if a feedwater isolation valve was closed during operation, the feedwater nozzle and spargers would undergo a severe thermal shock when feedwater was restored. This thermal shock could cause cracking and possible failure of the spargers and nozzles.

NRC Question No. 5

Describe the method utilized when exercising the excess flow check valves. (Reference Relief Request VR-8.)

Iowa Electric Response

During refueling outages, a flow path is established in the instrument line downstream of the excess flow check valves to verify the flow check valve closes. The valve handswitch is then operated to verify the excess flow check valve is open.

G. REACTOR RECIRCULATION

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valves MO-4627 and MO-4628 quarterly during power operation.

Iowa Electric Response

Technically, a quarterly test can be performed. However, such testing would require approximately 5 hours of operator action to place the plant in the proper configuration for testing. To prevent automatic runback of the pumps would require reduction of the pump speed to a minimum. This requires operator action to ensure the noise surveillance region of technical specifications is followed. To reduce core power, control rods would be inserted.

When closing the valve(s) the reactor level will increase. This creates the possibility of a reactor feedpump and main turbine trip which would result in a reactor scram. When opening the valve(s), reactor level decreases which could cause a scram due to low reactor water level. Additionally, when opening the valve, a sudden increase in core flow could cause a APRM flow-biased scram.

NRC Question No. 2

Provide a detailed technical justification for not full-stroke exercising valves MO-4629 and MO-4630 quarterly during power operation.

Iowa Electric Response

The above valves are currently open during power operation and could be exercised during power operation.

H. CONTROL ROD DRIVE HYDRAULIC

NRC Question No. 1

Provide the control rod drive scram testing Technical Specification acceptance criteria.

Iowa Electric Response

Technical Specification 3.3.C states:

1) The average scram insertion time, based on the deenergization of the scram pilot valve at time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Rod Position	Average Scram Insertion Times (SEC)
05	44	0.375
20	38	0.900
50	24	2.000
90	04	3.500

2) The average scram insertion times for the three fastest control rods of all groups of four control rods in a 2 X 2 array shall be no greater than:

% Inserted From Fully Withdrawn	Rod Position	Average Scram Insertion Times (SEC)
05	44	0.398
20	38	0.954
50	24	2.120
90	04	3.710

3) Maximum scram insertion time for 90% insertion of any operable control rod should not exceed 7.00 sec.

It is noted that by letter dated August 29, 1986 (NG-86-0112, RTS-192) Iowa Electric is proposing to change the rod scram time basis from a percentage insertion basis to a rod position basis to more accurately determine rod scram times.

NRC Question No. 2

Provide a detailed technical justification for not full-stroke exercising valves CV-1859A/B and CV-1867A/B quarterly during power operation.

Iowa Electric Response

To utilize the safety related control system to exercise these valves would require a manual reactor scram.

NRC Question No. 3

What is the safety function of valves SV-1851, SV-1852, SV-1853, and SV-1854?

Iowa Electric Response

There are 89 sets of these valves; one for each control rod drive. Normal insertion and withdrawal of the CRDs is accomplished by opening and closing a particular set of valves (only one CRD can be moved at a time). These valves are required to close or remain closed during a scram to allow the accumulator pressure to insert the control rod.

NRC Question No. 4

How are valves V-18-919 through V-18-1007 and V-18-118 through V-18-206 verified shut individually during testing?

Iowa Electric Response

Valves V-18-919 through V-18-1007 are verified shut during scram time testing. Additionally, weekly testing of the CRDs would detect a failure of these valves to close. (Reference Surveillance requirement 4.3.A.2.a.)

Valves V-18-118 through V-18-206 are verified shut during a pressure decay test.

I. RESIDUAL HEAT REMOVAL

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valve CV-1906 quarterly during power operation. Does this valve have a maximum stroke time assigned to it?

Iowa Electric Response

This valve serves as a high/low pressure interface. Exercising this valve during normal operation would place the plant in a degraded or unsafe condition by overpressurizing the low pressure side of the system. This valve does not have a stroke time associated with it. The valve is stroked in accordance with Paragraph IWV-3522, normally closed check valves and full flow tested to demonstrate operability.

NRC Question No. 2

Provide a detailed technical justification for not full-stroke exercising valves MO-1900 and MO-1901 quarterly during power operation.

Iowa Electric Response

These valves serve as a high/flow pressure interface. Exercising these valves during normal operation would place the plant in a degraded or unsafe condition because only one valve would remain to protect the low pressure portion of the line from overpressurization. In addition these valves are physically prohibited from opening unless reactor pressure is less than 135 psig.

NRC Question No. 3

Review the safety function of valves MO-1902 and MO-1903 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for these valves (penetration X-39B) is not a concern since the outboard valve is water sealed. Line leakage is a consideration, but line leakage does not meet the criteria for considering a valve to be Type A tested per Subparagraph IWV-2200(a). Valve MO-1902 is the subject of a separate relief request dated December 7, 1984 (copy included). We have been informally notified that approval is forthcoming.

NRC Question No. 4

Provide a detailed technical justification for not full-stroke exercising valves MO-1908 and MO-1909 quarterly during power operation.

Iowa Electric Response

These valves serve as a high/low pressure interface. Exercising these valves during normal operation would place the plant in a degraded or unsafe condition because only one valve would remain to protect the low pressure portion of the line from overpressurization. In addition, these valves are physically prohibited from opening unless reactor pressure is less than 135 psig.

NRC Question No. 5

Review the safety function of valves MO-1933, MO-1934, and MO-1935 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valves MO-1933 (penetration N-211A), MO-1934 (penetration N-210A) and MO-1935 (penetration N-210B) are not a concern since the valves are water sealed during accident conditions. Line leakage is a consideration for MO-1933, but does not meet the criteria for considering a valve to be categorized Type A per Subparagraph IWV-2200(a). Valve MO-1933 is the subject of a separate relief request dated December 7, 1984 (copy included). We have been informally notified that approval is forthcoming.

NRC Question No. 6

Review the safety function of valves MO-1949A/B to determine if they should be categorized A.

Iowa Electric Response

The RHR/Core Spray Fill pump (1P-70) maintains the RHR pressure greater than the maximum drywell accident pressure of 43 psi. In addition, this piping would remain water sealed during accident conditions. (The tail pipe is submerged in the suppression pool.) Therefore, any leakage past valves MO-1949A/B will be inleakage into the suppression pool.

NRC Question No. 7

Review the safety function of valves MO-1970 and MO-1989 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakages for valves MO-1970 (penetration N-210B) and MO-1989 (penetration N-225B) are not a concern since the valves are water sealed during accident conditions.

NRC Question No. 8

Review the safety function of relief valve PSV-1952 to determine if it should be categorized A/C.

Iowa Electric Response

The RHR/Core Spray Fill pump (1P-70) maintains the RHR pressure greater than the maximum drywell accident pressure of 43 psi. In addition, this piping would remain water sealed during accident conditions. (The tail pipe is submerged in the suppression pool.) Therefore, any leakage past valve PSV-1952 will be inleakage into the suppression pool.

NRC Question No. 9

How is the position of valves V-19-14 and V-19-16 individually verified during testing?

Iowa Electric Response

The valve(s) are verified closed by determining if the redundant pump attains reference values for flow and pressure. The valve(s) may be verified open by detection of local flow noise and proper operation of the pump.

NRC Question No. 10

Provide a detailed technical justification for not full-stroke exercising valve CV-2002 quarterly during power operation. Does this valve have a maximum stroke time assigned to it?

Iowa Electric Response

This valve serves as a high/low pressure interface. Exercising this valve during normal operation would place the plant in a degraded or unsafe condition by potentially overpressurizing the low pressure side. This valve does not have a stroke time associated with it. The valve is stroked at required flow to demonstrate operability.

NRC Question No. 11

Review the safety function of valves MO-2000 and MO-2001 to determine if they should be categorized A.

Iowa Electric Response

The NRC subcontractor's Technical Evaluation Report (TER) dated January 17, 1984 states that seat leakage for these valves (located on penetration X-39A) is not a concern since the outboard valve is water sealed. Line leakage is a consideration, but line leakage does not meet the criteria for considering a valve to be Type A tested per Subparagraph IWV-2200(a).

NRC Question No. 12

Review the safety function of valves MO-2006, MO-2007, and MO-2009 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valves MO-2006 (penetration N-211B), MO-2007 (penetration N-210B), and MO-2009 (penetration N-210A) are not a concern since the valves are water sealed during accident conditions. Line leakage is a consideration for MO-2006, but does not meet the criteria for considering a valve to be categorized Type A per Subparagraph IWV-2200(a). Valve MO-2006 is the subject of a separate relief request dated December 7, 1984. We have been informally notified that approval is forthcoming.

NRC Question No. 13

Review the safety function of valves MO-2038 and MO-2069 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valves MO-2038 (penetration N-210A) and MO-2069 (penetration N-225A) are not a concern since the valves are water sealed during accident conditions.

NRC Question No. 14

Review the safety function of valves MO-2044A/B to determine if they should be categorized A.

Iowa Electric Response

The RHR/Core Spray Fill pump (1P-70) maintains the RHR pressure greater than the maximum drywell accident pressure of 43 psi. In addition, this piping would remain water sealed during accident conditions. (The tail pipe is submerged in the suppression pool.) Therefore, any leakage past valves MO-2044A/B will be inleakage into the suppression pool.

NRC Question No. 15

Review the safety function of relief valve PSV-2043 to determine if they should be categorized A/C.

Iowa Electric Response

The RHR/Core Spray Fill pump (1P-70) maintains the RHR pressure greater than the maximum drywell accident pressure of 43 psi. In addition, this piping would remain water sealed during accident conditions. (The tail pipe is submerged in the suppression pool.) Therefore, any leakage past valve MO-2043 will be inleakage into the suppression pool.

NRC Question No. 16

How is the position of valves V-20-6 and V-20-8 individually verified during testing?

Iowa Electric Response

The valve(s) are verified closed by determining if the redundant pump attains reference values for flow and pressure. The valve(s) may be verified open by detection of local flow noise and proper operation of the pump.

NRC Question No. 17

Review the safety function of the following valves to determine if they should be included in the IST Program and categorized as indicated.

Category A/C		Category A	
PSV-1953	(C-4)	CV-1963	(D-3)
PSV-2042	(C-7)	CV-1964	(D-3)
		CV-2033	(D-7)
		CV-2034	(D-7)

Category B		Category C	
SV-1972	(C-3)	V-19-19	(C-6)
SV-1973	(C-3)	V-19-22	(C-6)
SV-2051	(C-7)	V-19-24	
SV-2052	(C-7)	V-19-124	
MO-2010	(D-5)		

Iowa Electric Response

MO-2010 - We will add to the IST Program as a Category B passive valve.
PSV-1953 - This valve is a 3/4" thermal relief and does not perform a function in shutting down the reactor. See also response to Question I.15.
PSV-2042 - See PSV-1953 above.
SV-1972 - This valve is a 1" valve to RHR sampling and does not perform a function in shutting down the reactor.
SV-1973 - See SV-1972 above.
SV-2051 - See SV-1972 above.
SV-2052 - See SV-1972 above.
CV-1963 - This valve would function in the RHR steam condensing mode only. The RHR steam condensing mode is not a safety-related function of RHR nor is it used at the DAEC. Per Article IWV-1000, only valves required to mitigate the consequences of an accident or required for safe shutdown of the reactor should be included in the IST Program.
CV-1964 - See CV-1963 above.
CV-2033 - See CV-1963 above.
CV-2034 - See CV-1963 above.
V-19-19 - This valve provides for keeping the RHR discharge line full to prevent water hammer during the starting of the RHR pumps. This function is not necessary to shutdown the reactor.
V-19-22 - This valve provides for keeping the Core Spray discharge line full to prevent water hammer during the starting of the Core Spray pumps. This function is not necessary to shutdown the reactor.
V-19-24 - See V-19-19 above.
V-19-124 - See V-19-19 above.

J. CORE SPRAY

NRC Question No. 1

Has the engineering evaluation concerning replacement or removal of the operators on valves CV-2118 and CV-2138 been completed? Relief Request VR-33 implies that the operators have been removed.

Iowa Electric Response

The review has been completed, and the operators are scheduled to be removed during the next refueling outage which will begin in mid-March 1987. The operators are presently disconnected from the power source.

NRC Question No. 2

Review the safety function of valves MO-2100, MO-2120, MO-2146, and MO-2147 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakages for valves MO-2100 (penetration N-227A), MO-2120 (penetration N-227B), MO-2146 (penetration N-227B), and MO-2147 (penetration N-227A) are not a concern since the valves are water sealed during accident conditions.

NRC Question No. 3

Review the safety function of valves MO-2104 and MO-2124 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakages for valves MO-2104 (penetration N-210A) and MO-2124 (penetration N-210B) are not a concern since the valves are water sealed during accident conditions.

NRC Question No. 4

Review the safety function of valves MO-2112 and MO-2132 to determine if they should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakages for valves MO-2112 (penetration N-210A) and MO-2132 (penetration N-210B) are not a concern since the valves are water sealed during accident conditions.

NRC Question No. 5

Review the safety function of valves PSV-2109, and PSV-2129 to determine if they should be categorized A/C.

Iowa Electric Response

The RHR/Core Spray Fill pump (1P-70) maintains the RHR pressure greater than the maximum drywell accident pressure of 43 psi. In addition, this line remains water sealed during accident conditions. (The tail pipe is submerged in the suppression pool.) Therefore, any leakage past valves PSV-2109 and PSV-2129 would be inleakage into the suppression pool.

NRC Question No. 6

How is the position of valves V-21-9 and V-21-12 individually verified during testing?

Iowa Electric Response

Proper operation of each core spray pump individually demonstrates that the check valves operate.

K. HPCI-STEAM SIDE

NRC Question No. 1

Would an entire safety system be rendered inoperable if valve MO-2238 failed while being tested? Should testing of this valve be done during cold shutdown?

Iowa Electric Response

Yes, MO-2238 is normally open and must remain open in order to operate the HPCI turbine. However, technical specifications require that this valve be cycled each month. (Reference Surveillance Requirement 4.5.D.1.c.)

NRC Question No. 2

Is valve V-22-16 equipped with an external operator? How is this valve exercised shut during cold shutdown?

Iowa Electric Response

No. Pressure is applied to the seat of the valve using the pressure decay method.

NRC Question No. 3

Does valve V-22-17 perform a safety function in the shut position?

Iowa Electric Response

Yes, the valve will be manually stroked closed during cold shutdown.

NRC Question No. 4

How is valve V-22-21 verified shut during cold shutdown?

Iowa Electric Response

Pressure is applied to the seat of the valve using the pressure decay method.

NRC Question No. 5

Does valve V-22-22 perform a safety function in the shut position?

Iowa Electric Response

Yes, the valve will be manually stroked closed during cold shutdown.

NRC Question No. 6

Can valves V-22-63 and V-22-64 be verified shut during power operation?

Iowa Electric Response

No. Verifying the valves shut would require HPCI to be inoperable.

NRC Question No. 7

Is valve PSV-2290 in service as a vacuum breaker? Should this valve be included in the IST Program?

Iowa Electric Response

No. This valve has been capped. The safety related vacuum breaker for the system are valves V-22-64 and V-22-63.

NRC Question No. 8

Review the safety function of valve CV-2234 to determine if it should be included in the IST Program.

Iowa Electric Response

We will add CV-2234 to the IST Program as a Category B valve and delete CV-2235 since only single valve isolation is needed for this line.

L. HPCI - WATER SIDE

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valve CV-2313 quarterly during power operation. Should this valve be categorized A/C?

Iowa Electric Response

This valve is equipped with an operator that cannot be cycled with any pressure drop across the valve. The valve serves no containment isolation function since it is a simple check valve. MO-2312 is the containment isolation valve for that line.

NRC Question No. 2

Review the safety function of valve MO-2318 to determine if it should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valve MO-2318 (penetration N-210A) is not a concern since the valve is water sealed during accident conditions.

NRC Question No. 3

Review the safety function of valve MO-2321 to determine if it should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valve MO-2321 (penetration N-226) is not a concern since the valve is water sealed during accident conditions.

NRC Question No. 4

Does valve V-23-4 perform a safety function in the closed position while the HPCI suction valves shift to align the pump suction to the suppression pool?

Iowa Electric Response

This valve does not perform a safety function when shifting suction to the suppression pool. V-23-001 is the valve that performs a safety function when shifting suction from the CST to the suppression pool. Refer to Relief Request VR-21.

NRC Question No. 5

Review the safety function of valves MO-2315 and V-23-14 to determine if they should be included in the IST Program and categorized B and C, respectively.

Iowa Electric Response

MO-2315 (recently changed to CV-2315) will be added to the IST Program as a category B valve. V-23-14 is already included in the IST Program as a category C valve.

M. RCIC-STEAM SIDE

NRC Question No. 1

Would an entire safety system be rendered inoperable if valve MO-2400 failed while being tested? Should testing of this valve be done during cold shutdown?

Iowa Electric Response

Yes, MO-2400 is normally open and must remain open in order to operate the RCIC turbine. However, technical specifications require that this valve be cycled each month. (Reference Surveillance Requirement 4.5.E.1.c.) It should also be noted that RCIC is not considered a safety related system.

NRC Question No. 2

Does valve V-24-8 perform a safety function in the shut position? How is this valve full-stroke exercised?

Iowa Electric Response

Yes, The RCIC system is not considered to be a safety related system; however, the valve will be manually stroked closed during cold shutdown. The valve is stroked open by RCIC turbine exhaust.

NRC Question No. 3

Is valve V-24-23 equipped with an external operator? How is this valve verified shut during cold shutdowns?

Iowa Electric Response

No. Pressure is applied to the seat of the valve by the pressure decay method.

NRC Question No. 4

How are valves V-24-46 and V-24-47 verified shut during cold shutdown?

Iowa Electric Response

The valves are verified shut by applying pressure to the seat of each valve by the pressure decay method.

NRC Question No. 5

Review the safety function of the following valves to determine if they should be included in the IST Program and categorized as indicated.

Category B	Category C
CV-2435	V-24-9 V-24-10

Iowa Electric Response

We will add CV-2435 to the IST Program as a Category B valve and delete CV-2436 since only single valve isolation is needed for this line. Valves V-24-09 and V-24-10 do not perform a function in shutting down the reactor and they are not safety related as RCIC is not considered a safety related system.

NRC Question No. 6

Do valves PCV-2414 and PCV-2427 have a required fail-safe position? If so, they should be included in the IST Program and tested in accordance with Section XI.

Iowa Electric Response

No. These valves are pressure regulating only and therefore are exempt from the requirements of IWV-1200(a).

N. RCIC-WATER SIDE

NRC Question No. 1

Provide a detailed technical justification for not full-stroke exercising valve CV-2513 quarterly during power operation. Should this valve be categorized A/C?

Iowa Electric Response

This valve is equipped with an operator that cannot be cycled with any pressure drop across the valve. In accordance with General Design Criteria 55 of Appendix A to 10 CFR Part 50, the valve serves no containment isolation function since it is a simple check valve. MO-2512 is the containment isolation valve for that line.

NRC Question No. 2

Review the safety function of valve MO-2510 to determine if it should be categorized A.

Iowa Electric Response

On page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valve MO-2510 (penetration N-210A) is not a concern since the valve is water sealed during accident conditions.

NRC Question No. 3

Review the safety function of valve MO-2516 to determine if it should be categorized A.

Iowa Electric Response

on page 3 of the NRC Safety Evaluation Report (SER) dated January 17, 1984, the NRC staff agreed with the subcontractor's Technical Evaluation Report (TER) that seat leakage for valve MO-2516 (penetration N-224) is not a concern since the valve is water sealed during accident conditions.

NRC Question No. 4

Review the safety function of valve V-25-03 to determine if it should be included in the IST Program. Does this valve perform a safety function in the closed position while the RCIC suction valves shift to align the pump suction to the suppression pool?

Iowa Electric Response

This valve does not perform a safety function when shifting suction to the suppression pool. V-25-001 is the valve designed to prevent backflow into the suppression pool in the event of pump suction shift from the CST to the suppression pool. Refer to Relief Request VR-21. It should be noted that RCIC is not considered to be a safety related system.

NRC Question No. 5

Review the safety function of valve MO-2515 to determine if it should be included in the IST Program and categorized B.

Iowa Electric Response

MO-2515 will be added to the IST Program and categorized B.

O. COMPRESSED AIR

NRC Question No. 1

Is the blind flange installed on the breathing air line at penetration 21? If it is not installed, then should valve V-30-288 be included in the IST Program and Categorized A, passive?

Iowa Electric Response

Yes, the blind flange is installed.

P. DIESEL GENERATOR SYSTEMS

NRC Question No. 1

Review the safety function of the following check valves to determine if they should be included in the IST Program and tested in accordance with Section XI.

V-32-19	V-32-45
V-32-21	V-32-52
V-32-43	V-32-54

Iowa Electric Response

We subscribe to the ASME Section XI position that the intent of Subarticle IWV-1100 does not pertain to systems containing medium other than steam or water. (See the attached ASME response dated February 16, 1978.)

NRC Question No. 2

Are the emergency diesel engines equipped with air start solenoids? If so, how many are installed on each engine and can they be tested individually?

Iowa Electric Response

Yes, valves SV-3261A, SV-3261B, SV-3262A and SV-3262B are diesel air start solenoids. There are two air start solenoids for each diesel which can be tested individually. We subscribe to the ASME Section XI position that the intent of Subarticle IWV-1100 does not pertain to systems containing medium other than steam or water. (See the attached ASME response dated February 16, 1978.)

Q. CONTAINMENT ATMOSPHERE CONTROL

NRC Question No. 1

Why were valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4304, CV-4305, and CV-4306 deleted from Revision 7 of the IST Program?

Iowa Electric Response

The page was inadvertently omitted from your copy of the submittal, as the valves have not been deleted from Revision 7. The missing page is included in this submittal.

NRC Question No. 2

Provide a detailed technical justification for not full-stroke exercising valves V-43-82, V-43-84, V-43-86, and V-43-88 quarterly during power operation and cold shutdowns.

Iowa Electric Response

Injection of nitrogen would cause pressurization of the containment resulting in unnecessary safety system actuations as the only means to test these valves is by actual injection. Injection of nitrogen would place the plant in a Limiting Condition for Operation (LCO). Also, the containment atmosphere is not necessarily purged of nitrogen every cold shutdown. Refer to Technical Specification 3.7.A.6.b. Relief Request VR-24 proposes alternate testing to that which is required by Code.

NRC Question No. 3

Do valves PCV-4320A/B and PCV-4323A/B have a required fail-safe position? If so, they should be included in the IST Program and tested in accordance with Section XI.

Iowa Electric Response

No. These valves are pressure regulating only and therefore are exempt from the requirements of IWV per IWV-1200(a).

R. DRYWELL COOLING WATER

NRC Question No. 1

Review the safety function of valves V-57-58 and V-57-59 to determine if they should be included in the IST Program and categorized C.

Iowa Electric Response

The lines associated with these valves do not perform a safety function.

S. MSIV LEAKAGE CONTROL

NRC Question No. 1

Review the safety function of valves MO-8401A, B, C, and D to determine if they should be categorized A.

Iowa Electric Response

Leakage through these valves is not a concern as any leakage would be processed through either the standby gas treatment system or closed radwaste system. The system is designed to pass flow following a postulated accident.

PUMP TESTING PROGRAM

AA. EMERGENCY SERVICE WATER

NRC Question No. 1

IWP-1200(a) excludes pump drivers from the requirements of Section XI unless the pump and driver are an integral unit and the pump bearings are located in the driver, therefore, is Relief Request PR-2 necessary? Are the emergency service water pumps submerged? How are pump vibration readings taken if the pumps are submerged and inaccessible?

Iowa Electric Response

The pumps in question are vertical lineshaft pumps. The bearings are located on the line shaft and are inaccessible; however, the pump is accessible and the pump vibration readings can be taken near the pump motor mount flange.

BB. SCREEN WASH

NRC Question No.1

Has the evaluation of the instrumentation requirements for the screen wash pumps been completed? The current NRC position is that lack of installed instrumentation is not an acceptable long term technical justification for not measuring the Code required parameters on pumps that perform a safety-related function.

Iowa Electric Response

Yes, the evaluation is complete and instrumentation has been installed to meet the requirements for testing the Screen Wash pumps. The relief request can now be voided.

CC. RIVER WATER

NRC Question No. 1

How are vibration readings taken if the pumps are submerged and inaccessible?

Iowa Electric Response

The pumps in question are vertical lineshaft pumps. The bearings are located on the line shaft and are inaccessible; however, the pump is accessible and the pump vibration readings can be taken near the pump motor mount flange.

DD. HPCI AND RCIC

NRC Question No. 1

Has the evaluation of the instrumentation requirements for the HPCI and RCIC pumps been completed? The current NRC position is that lack of installed instrumentation is not an acceptable long term technical justification for not measuring the Code required parameters on pumps that perform a safety-related function.

Iowa Electric Response

Yes, the evaluation is completed and instrumentation has been installed to meet the requirements for testing the HPCI and RCIC pumps. The relief request can now be voided.

EE. DIESEL FUEL OIL

NRC Question No. 1

In reference to Relief Request PR-10, IWP-3320(d) allows instrument recalibration and retesting if the results of the previous pump test fall outside the allowable ranges of Table IWP-3100-2.

Iowa Electric Response

We agree.

FF. MISCELLANEOUS SYSTEMS

NRC Question No. 1

In reference to Relief Request PR-5, why is it more difficult to duplicate reference flow rates than it is to duplicate a slightly higher and slightly lower flow rate during pump testing?

Iowa Electric Response

In order to duplicate exactly one flow rate and differential pressure, the throttling valve will have to be adjusted slightly more open, then slightly more closed, until the exact point on the pump curve can be duplicated. The valves used in the plant are not designed for precise throttling, but rather, only for demonstrations that the pump can meet its designed criteria. Such valve manipulation can damage valve operator components and valve internals, unnecessarily degrading the valve.

NRC Question No. 2

As an alternate to pump vibration amplitude measurement required by IWP-3100, the NRC currently accepts the measurement of vibration velocity using the General Machinery Vibration Severity Chart as criteria for acceptable velocities. Provide specific justification for defining as acceptable any velocity values greater than 0.314 in./sec which are considered to be ROUGH vibration levels on the chart. (Refer to pump Relief Request #8.)

Iowa Electric Response

The relief request will be withdrawn and vibration measurements will be taken per IWP-3100.

NRC Question No. 3

Review the safety function of the spent fuel pit cooling pumps to determine if they should be included in the IST Program and tested in accordance with Section XI.

Iowa Electric Response

No, the spent fuel pit cooling pumps are not safety-related. In addition to the makeup capabilities of the RHR systems, emergency makeup and cooling is provided by a manual hose connection to the Emergency Service Water System (UFSAR 9.1.3.3).

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

January 17, 1984

RECEIVED
JAN 26 1984
I. E. L. P. CO.
By

Docket No. 50-331

RECEIVED
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JAN 26 1984
NUCLEAR LICENSING
and FUELS

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

Re: Duane Arnold Energy Center

The Commission has issued the enclosed Exemption from certain requirements of Section 50.54(o) and Appendix J to 10 CFR Part 50 for the Duane Arnold Energy Center, in response to your letter dated August 29, 1978 as supplemented by letter dated November 5, 1981 and clarified through telephone discussions with the staff on October 1, 1982. This Exemption, which is being forwarded to the Office of the Federal Register for publication, permits the testing of main steam isolation valves at a pressure of 24 psig, and extends the interval between Type B tests for the containment airlock doors at accident pressure (Pa).

Your request, however, to exempt core spray isolation valves and RCIC and HPCI condensate return isolation valves from Type C testing has been denied. Furthermore, we have evaluated your request for exemptions related to certain other lines and valves meeting various specific requirements as described in the enclosed Safety Evaluation, and have determined that exemptions for these items are not necessary.

The bases for our findings and the disposition of all of the exemption requests are contained in the enclosed Safety Evaluation.

Within 60 days of the date of this letter please propose Technical Specifications reflecting the Appendix J testing requirements based on this Exemption.

Sincerely,

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Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Exemption
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Lee Liu
Iowa Electric Light and Power Company
Duane Arnold Energy Center

cc:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	Docket No. 50-331
IOWA ELECTRIC LIGHT AND POWER COMPANY	)	
(Duane Arnold Energy Center)	)	

EXEMPTION

I.

The Iowa Electric Light and Power Company (IELP/the licensee) is the holder of Facility Operating License No. DPR-49 (the license) which authorizes operation of the Duane Arnold Energy Center (DAEC) located in Linn County, Iowa, at steady state reactor core power levels not in excess of 1658 megawatts thermal. This license provides, among other things, that it is subject to all rules, regulations and Orders of the Commission now or hereafter in effect.

II.

Section 50.54(o) of 10 CFR Part 50 requires that primary reactor containments for water cooled power reactors be subject to the requirements of Appendix J to 10 CFR Part 50. Appendix J contains the leakage test requirements, schedules, and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. Appendix J was published on February 14, 1973 and in August 1975, each licensee was requested to review the extent to which its facility met the requirements.

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On August 7, 1975, IELP submitted its evaluation of the DAEC in which it assessed compliance with the rule and also requested an exemption from certain requirements of the rule. The IELP submittal for the DAEC was supplemented by letter dated August 29, 1978 and November 5, 1981 and clarified in a telephone discussion on October 1, 1982. In these submittals, IELP requested that certain test methodology, components, and penetrations be exempted from Appendix J requirements. The Franklin Research Center, as a consultant to NRR, has reviewed the licensee's submittals and prepared a Technical Evaluation Report (TER) dated March 17, 1982. The NRC staff has reviewed this TER, and in its Safety Evaluation dated April 2, 1982, concurred in the TER's bases and findings. However, for Item 2 below, pertaining to airlock door testing, the staff performed an additional evaluation prior to determining the acceptability of the licensee's request.

1. Section III.C.2 of Appendix J requires, in part, that Type C testing be performed at the peak calculated accident pressure (Pa). IELP requested an exemption from this requirement for the Main Steam Isolation Valves (MSIVs) to permit testing at 24 psig rather than at Pa (48 psig) and submitted certain design information as justification.

The MSIVs are leak tested by pressurizing between the valves. The MSIVs are angled in the main steam lines in the direction of flow in order to afford better sealing upon closure. A test pressure of Pa acting under the inboard disc is sufficient to lift the disc off its seats, and results in excessive leakage into the reactor vessel. This would result in a meaningless test. The proposed test calls for a test pressure of 24 psig to avoid lifting the disc at the inboard valve. The total observed leakage through both valves (inboard and outboard) is then conservatively assigned to the penetration. On this basis, we conclude that testing at a reduced pressure of 24 psig is acceptable.

2. In a letter dated November 5, 1981, IELP requested an exemption from the airlock door testing requirements of Section III.D.2(b), which was revised effective October 22, 1980. The revised rule required testing of the airlocks as follows:

   a. Every six months at a pressure of not less than Pa (and after periods when the airlock is opened and containment integrity is not required).

   b. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, at a pressure of Pa or at a reduced pressure as stated in the Technical Specifications.

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's proposal to (1) test containment airlocks at a pressure of Pa and at an interval not longer than one operating cycle, and (2) whenever the airlock was opened during the operating cycle, and containment integrity was required, the airlock gasket would be tested at Pa following closure if it had been greater than 3 days since the last leakage test.

FRC concluded that the licensee's proposal to test airlock gaskets within 3 days of an airlock opening is acceptable. However, FRC did not find acceptable the licensee's proposal to test the entire airlock at a pressure of Pa once per operating cycle, since it did not make adequate allowances to detect potential deterioration of airlocks through normal use, to detect possible damage to the door mechanism, to detect potential damage to door seals through moving equipment into and out of containment, and to detect possible fouling of seals during closure. FRC proposed that testing of the entire airlock assembly at a pressure of Pa should be conducted at the six-month interval as required by Appendix J.

We agree with the FRC's conclusion that the airlock gasket leakage be tested within 3 days from an airlock opening. We further agree with the FRC's conclusion that the airlock testing frequency should make adequate allowances to detect potential deterioration of airlocks through normal use. However, when the airlock remains closed, that is, there is no opening or closing of the doors to cause degradation of seals or damage to door mechanisms, we find that the reduced pressure testing frequency proposed by the licensee would be adequate to assure that the airlock door seal integrity is maintained.

Based on the above, the staff has reevaluated the six-month test requirement and has developed a revised position which meets the objectives of Appendix J requirements for containment airlock door tests. This revised position still requires the containment airlock to be tested at six-month intervals at a pressure of Pa in accordance with Appendix J, except that this test interval may be extended up to the next refueling outage (up to a maximum interval between Pa tests of 24 months) if there have been no airlock openings since the last successful test at Pa. The intent of the Appendix J requirement is to assure that the airlock door seal integrity is maintained and that no degradation has occurred as a result of opening of the airlock doors between testing intervals at Pa. This position satisfies the objectives of the requirement. The licensee has proposed that the personnel airlock be pressurized to Pa and leak-tested at an interval no longer than one operating cycle (up to a maximum interval between Pa tests of 24 months). We find this consistent with our position and therefore acceptable, except that the six-month testing interval is still applicable if the containment airlock door has been opened since the last successful test at Pa.

The licensee will be requested to propose appropriate modifications to the Technical Specifications.

III.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, the Commission hereby approves the following exemption requests:

1. Exemption is granted from the requirements of Section III.C.2 of Appendix J pertaining to the Type C testing of the main steamline isolation valves at a test pressure of Pa to the extent that testing is to be conducted at pressure Pa. Testing at a reduced pressure of 24 psig is acceptable due to the unique design of the valves.
2. Exemption is granted from the requirements of Section III.D.2 of Appendix J pertaining to the test frequency for conducting Type B tests at six-month intervals at a test pressure of not less than Pa to the extent that the testing is to be conducted at six-month intervals after initial fuel loading. The test interval may be

extended beyond the six-month test interval to the next refueling outage, but in no case shall exceed 24 months from the last test at Pa, provided that there have been no airlock openings since the last successful test at Pa.

The NRC staff has determined that the granting of this exemption will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with this action.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 17th day of January, 1984.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

APPENDIX J REVIEW

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 Introduction

On August 7, 1975 (Reference 1), the NRC requested Iowa Electric Light and Power Company (IELP/licensee) to review its containment leakage testing program for Duane Arnold Energy Center (DAEC) and the associated Technical Specifications, for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since by this date there were already many operating nuclear plants and a number more in advanced stages of design or construction, the NRC decided to have these plants reevaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the licensee for DAEC. The results of our evaluation are provided below.

2.0 Evaluation

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's submittals (References 2 and 3) and prepared the enclosed Technical Evaluation Report (TER-C5257-13), Containment Leakage Rate Testing for Duane Arnold Energy Center. We have reviewed FRC's evaluation and concur in its bases and findings, with the exception of its assessment of the licensee's request for exemption pertaining to the frequency of Type B tests for the containment airlock, which is further evaluated below.

Section III.D.2 of Appendix J, effective October 22, 1980, requires testing of the airlock as follows:

1. Every six months at a pressure of not less than accident pressure (Pa) and after periods when the airlock is opened and containment integrity is not required.

2. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, at a pressure of Pa or at a reduced pressure as stated in ~~the~~ Technical Specifications.

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By letter dated August 9, 1978, the licensee requested an exemption from the frequency requirements of Section III.D.2 in order to permit testing on a frequency consistent with the plant operating cycle (i.e., each refueling outage). FRC's evaluation of the licensee's submittals in support of the exemption request which is contained in the enclosed TER concluded that the licensee's program related to the test frequency and pressure should conform to the requirements of Section III.D.2 of Appendix J.

However, subsequent discussions with the licensee regarding test methodology and additional evaluation by the staff of airlock degradation causal factors and operating history have resulted in a reevaluation of our position. Test performance requires shutting down the reactor and opening the equipment hatch in order to install a strongback on the inner airlock door to prevent unseating the airlock door, and subsequent door and hatch openings to remove the strongback. This would result in an outage of several days for the licensee, the cost of replacmeent power to the public, and could subject operating personnel to additional radiation exposure. In addition, the additional openings of the equipment hatch and airlock provide additional opportunities for inadvertent seal degradation.

Based on these considerations, we have developed the following modified position which we believe meets the objectives of Appendix J requirements for Type B tests of containment airlocks.

We will still require containment airlocks to be tested every six months at a pressure of not less that Pa in accordance with Appendix J, except that the test interval may be extended to the next refueling outage (up to a maximum interval between Pa tests of 24 months) provided that there have been no airlock openings since the last successful test at Pa and a Pa test is performed following the next airlock opening. The intent of the Appendix J requirement is to assure that the airlock door seal integrity is maintained and no degradation has occureed as a result of opening of the airlock doors between testing intervals at Pa. Since there is an inadequate basis to conclude that no airlock seal degradation occurs if the airlock doors have not been opened between extended testing invervals at Pa, we believe that a reduced pressure testing or testing between seals every six months should be performed to assure that the airlock door seal integrity is maintained between the extended testing intervals at Pa. We believe this position satisfies the objectives of the requirements. The licensee will be requested to propose appropriate modifications to his Technical Specifications.

Therefore, the exemption from the airlock testing frequency requirements of Appendix J requested by the licensee should be granted provided the licensee complies with the staff's revised position on airlock testing.

3.0 Conclusion

Based on our review of the enclosed technical evaluation report regarding the October 13, 1975, August 9, 1978, and May 9, 1980 Appendix J submittals by the licensee for DAEC, we conclude the following:

3.1 Potential Exemptions from Appendix J (Reference 2)

No exemption from Appendix J is required for penetrations X-9A and X-9B as a result of the licensee's commitment to modify the inboard feedwater isolation valves.

Deletion of RHR Shutdown cooling supply valves MO-1908 and MO-1909 (penetration X-12) from Type C testing is acceptable because Appendix J does not require testing of these valves. Therefore, no exemption is required.

Type C testing of core spray isolation valves MO-2115, MO-2117, MO-2135, and MO-2137 is required unless testing of the core spray system demonstrates that the first isolation valve remains water covered throughout the post-accident period. One of the licensee's submittals (Reference 2) proposed capping penetration X-36 on both sides of the penetration so isolation valves V-17-52, V-17-53 and V-17-54 may be deleted from Type C testing. The licensee has since decided not to cap penetration X-36 and committed to perform Type C testing on the isolation valves associated with this penetration. Therefore, no exemption is required.

The licensee's proposal to delete RCIC and HPCI condensate return isolation valves from Type C testing is unacceptable because the valves are relied upon to perform a containment isolation function (i.e., isolate a direct path to the atmosphere from the main steam system of a BWR) when the RCIC or HPCI systems are in operation after an accident. Valves CV-2410, CV-2411, CV-2211, and CV-2212 should continue to be Type C tested. Therefore, this exemption request is denied.

Main steam isolation valves may continue to be tested at 24 psig because the test will provide a conservative measure of the leakage exiting at a pressure of Pa due to the design of the valves. The proposed exemption from the Appendix J requirement to test these valves at Pa is acceptable. Type C testing is not required and no exemption is necessary for the following penetrations because Appendix J does not require testing: N-210A & B, N-224, N-225A & B, N-226, N-227A & B, X-13A & B and X-17. For penetration X-39B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard isolation valves in order to test the valve packing and body-to-bonnet seals of the inboard valve. For penetration N-211A & B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard valves provided that this testing will expose the packing and body-to-bonnet seal areas of the inboard valves to the test pressure.

The licensee's proposal to test the RCIC and HPCI turbine exhaust return lines to the suppression pool (penetrations N-212, N-214, N-222) with water and to add the results of the air leakage totals for compliance with technical specifications limits is acceptable. Therefore, no exemption is required.

The Franklin Research Center concluded that a full containment airlock test at a pressure of Pa once every six months is required and that the licensee's proposal to conduct this testing once every operating cycle is unacceptable. The staff has however, reevaluated the airlock testing requirement as discussed in Section 2.0 of this Safety Evaluation. The staff now agrees with the licensee that without this exemption from Appendix J requirements, the plant would have to be shut down and the equipment hatch opened to install a strongback on the inner airlock door to perform the test and subsequent door and hatch opening to remove the strongback. This would result in an outage of several days for the licensee, the cost of replacement power to the public, and could subject the operating personnel to additional radiation exposures. In addition, the additional openings of the equipment hatch and airlock provide additional opportunities for inadvertent seal degradation. The staff has, therefore, revised its position to permit the airlock testing interval to extend up to next refueling outage if there have been no airlock openings since last successful test at Pa.

Testing of airlock gaskets at a pressure of Pa within three days of airlock opening is acceptable. No exemption is required.

3.2 Proposed Changes to the Technical Specifications (Reference 3)

Note 2 of Table 3.7-1 regarding the testing of containment airlocks should be changed to read "To be tested at least once every six months" in lieu of "To be tested at least each operating cycle." The staff has, however, reevaluated this position as discussed in Section 2.0.

The addition of a flange "O"-ring to penetration 213 in Table 3.7-1 is acceptable.

The deletion of valves V-14-2, V-14-4, V-17-80, V-17-84, and V-22-60 from Table 3.7-2 is acceptable because Appendix J does not require that they be tested. Valves CV-2410, CV-2411, CV-2211, and CV-2212 should not be deleted from Table 3.7-2.

Deletion of valves MO-1908 and MO-1909 from Table 3.7-2 is acceptable because Appendix J does not require that they be tested. Valves MO-2115, MO-2117, MO-2135 and MO-2137 should not be deleted from Table 3.7-2 unless the licensee's testing of the core spray system is used to demonstrate a water seal on the isolation valves throughout the post-accident period.

The deletion from Table 3.7-2 of 10 inaccessible, normally open manual valves in closed systems inside containment is acceptable because only the outside valves are relied upon as containment isolation valves in accordance with GDC 57.

Valves V-17-54, V-17-52, and V-17-53 should not be deleted from Table 3.7-2 because the associated penetration is not being deleted.

Testing of valves in the direction opposite the pressure existing in the post-accident condition is acceptable but the licensee should retain onsite documentation of the determination that the reverse-direction testing is equivalent or more conservative than testing in the direction of post-accident pressure.

For penetrations provided with a pressurization system, the proposed changes to the Technical Specifications should be modified to include the three years limitation between testing.

Other miscellaneous changes were found acceptable as discussed in Table 3-1 of the enclosed FRC report dated March 17, 1982.

References

1. K. Goller, Assistant Director for Operating Reactors letter to IELP; dated August 7, 1975.

2. Lee Liu, Vice President, IELP letter to K. Goller, Assistant Director for Operating Reactors, dated October, 13, 1975.

3. Lee Liu, IELP, "IELP Application for Amendment of DPR-49 and the Technical Specifications" to H. Denton, Director, Office of Nuclear Reactor Regulation, dated August 9, 1978.

4. L. Root, Assistant Vice President, IELP letter to T. Ippolito, Chief, ORB#3, dated May 9, 1980.

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Enclosure:
Technical Evaluation Report

Dated: January 17, 1984

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

IOWA ELECTRIC LIGHT AND POWER COMPANY
DUANE ARNOLD ENERGY CENTER UNIT 1

NRC DOCKET NO. 50-331

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FRC PROJECT C5257

FRC ASSIGNMENT 1

FRC TASK 17

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March 17, 1982

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. T. J. DelGaizo contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. BACKGROUND

On August 7, 1975 [1], the Nuclear Regulatory Commission (NRC) requested Iowa Electric Light and Power Company (IEL) to review its containment leakage testing program for Duane Arnold Energy Center Unit 1 (DAEC) and to provide a plan for achieving full compliance with 10CFR50, Appendix J, where necessary. The review was to include appropriate design modifications, changes to technical specifications, and requests for exemption from the requirements pursuant to 10CFR50.12.

IEL replied on October 13, 1975 [2], listing several areas where differences existed between the current technical specifications at DAEC and 10CFR50, Appendix J. IEL further stated that the apparent differences would be reviewed prior to proposing technical specification changes or requests for exemption from the regulation. Following an exchange of correspondence with the NRC, IEL submitted an Application for Amendment of DPR-49 on August 29, 1978 [3]. This letter responded to an NRC request for additional information relative to the differences identified in Reference 2, provided technical specifications changes for DAEC reflecting these responses, and proposed additional changes along with supporting rationale.

The purpose of this report is to provide technical evaluations of all outstanding issues pertaining to the implementation of 10CFR50, Appendix J, at DAEC. Consequently, it provides technical evaluations of the potential exemptions from the requirements of Appendix J submitted by Reference 2 and amplified in Reference 3 and also provides technical evaluations of the proposed changes to the technical specifications submitted in Reference 3.

2. EVALUATION CRITERIA

Code of Federal Regulations, Title 10, Part 50 (10CFR50), Appendix J, Containment Leakage Testing, was the criteria for the evaluation of these submittals. Furthermore, in recognition of plant-specific conditions which could lead to a request for exemption not explicitly covered by the regulation, the NRC directed that technical reviews constantly emphasize the basic intent of Appendix J, that potential containment atmospheric leakage paths be identified, monitored, and maintained below established limits.

3. TECHNICAL EVALUATION

3.1 EXEMPTIONS FROM THE REQUIREMENTS OF APPENDIX J

In Reference 2, IEL identified several areas where differences existed between the current technical specifications at DAEC and 10CFR50, Appendix J. Reference 3 provided additional information related to these differences. Each of these potential exemptions from the requirements of Appendix J is evaluated in the following paragraphs.

3.1.1 Local Leak Rate Testing of Isolation Valves

3.1.1.1 Feedwater, HPCI, and RCIC Injection Isolation Valves (Penetrations X-9A and X-9B)

In Reference 2, IEL proposed to continue testing the valves associated with the isolation of penetrations X-9A and X-9B with water in lieu of air (valves V-14-1, MO-4442, MO-2512, MO-2740, V-14-3, MO-4441, and MO-2312). In Reference 3, however, IEL committed to replace the inboard feedwater isolation valves by the end of the 1980 refueling outage with valves capable of being air-tested. IEL stated that, because of this modification, valves V-14-1, V-14-1, MO-4442, MO-2512, MO-2740, V-14-3, MO-4441, and MO-2312 will be air-tested.

Evaluation

Based upon IEL's commitment to modify the inboard feedwater isolation valves, there is no longer a need for an exemption for penetrations X-9A and X-9B because the Type C testing requirements of Appendix J will be met. IEL's plan to modify the valves by the end of the 1980 refueling outage is acceptable, and therefore, no further evaluation is required regarding these valves.

3.1.1.2 RHR Shutdown Cooling Supply (Penetration X-12)

In Reference 3, IEL stated that RHR shutdown cooling supply valves MO-1908 and MO-1909, associated with penetration X-12, should be deleted from

Type C testing requirements since these valves do not meet any of the containment isolation valve criteria as listed in Section II.H of Appendix J. IEL further stated that, since all containment boundaries are passive, except for the pumps which are redundant, no single active failure will cause a loss of the containment function.

Evaluation

Sections II.H and III.A.1(d) of Appendix J identify the containment isolation valves which may require Type C testing. Furthermore, Section II.H defines containment isolation valves as those valves which are relied upon to perform a containment isolation function.

The RHR system is designed to engineered-safety-feature-system standards to ensure that it will remain operational and water filled throughout the period following a postulated LOCA. IEL has stated, and FRC concurs, that there is no single active failure which will cause a loss of the containment function. Therefore, there is no potential for leakage of containment atmosphere through penetration X-12, and valves MO-1908 and MO-1909 are not relied upon to perform a containment function.

Consequently, deletion of these valves from Type C testing is acceptable because Appendix J does not require testing. No exemption from Appendix J is required.

3.1.1.3 Core Spray Pump Discharge Valves (Penetrations X-16A and X-16B)

In Reference 3, IEL proposed to delete core spray pump discharge valves MO-2115, MO-2117, MO-2135, and MO-2137 from the list of valves to be Type C tested because that the core spray system is a seismic Class I system and that "the core spray system external to the containment is the second boundary whose integrity is proven periodically during system operational checks."

In Reference 4, IEL provided additional information relative to the system operational checks of the core spray system. IEL reported that the system operational checks have now become part of the "Integrity of Systems Outside Containment" tests that are conducted each refueling cycle to meet the requirements of NUREG-0578 as developed by the BWR Owner's Group. For the core spray system, IEL reported that tests are performed quarterly at a minimum pressure of 113 psig (Pa at DAEC is 54 psig). The tests are performed under a preventive maintenance program designed to maintain system leakage as low as practical, with inspections being performed in conjunction with the system pressure tests required by Section XI of the ASME Boiler and Pressure Vessel Code.

Evaluation

The core spray system is a two-independent-loop system, each loop containing a single pump. Under expected post-accident conditions, there is no possibility of leakage of containment atmosphere through this system because the system will be operating with a water pressure higher than peak containment accident pressure. However, should one of the pumps fail to start under accident conditions, containment atmosphere would enter the system and the system outside containment would become a potential path for the leakage of air beyond the containment boundary.

IEL proposes to delete the four motor-operated isolation valves located outside containment (two in series in each loop) from the list of valves to be Type C tested. IEL's position is that the core spray system external to the containment provides the leakage boundary and that this boundary is tested quarterly. The testing is performed at a minimum of 113 psig with an acceptance criterion requiring as-low-as-practical leakage. The system is a seismic Class I system and is designed to remain intact following a postulated accident.

However, in order to demonstrate that the containment isolation valves of the core spray system are not relied upon to perform a containment isolation function, it is necessary to demonstrate that the valves remain water sealed throughout the post-accident period. Therefore, the periodic test of the system outside containment would need to actually measure an integrated system liquid leakage rate and compare the measured rate with that leakage rate which will just exhaust the available water inventory inside containment between the area of the break and the first isolation valve outside containment during the period when the containment is pressurized following the accident. If the measured integrated system leakage rate is lower than the calculated rate, the test would demonstrate that the first isolation valve outside containment would remain water sealed throughout the post-accident period. In this condition, the isolation valve is not relied upon to prevent the escape of containment air to outside atmosphere throughout the post-accident period; therefore, the valve does not qualify as a containment isolation valve in accordance with Section II.B of Appendix J and does not require Type C testing.

Unless actual testing demonstrates that the first isolation valve remains water covered throughout the post-accident period (demonstrated with the periodicity of the Type C tests), there is no technical basis for determining that the isolation valve is not relied upon to perform a containment isolation function in accordance with Appendix J. Therefore, Type C testing of the containment isolation valves is required.

3.1.1.4 CRD Return Line (Penetration X-36)

In Reference 2, IEL proposed to test valves V-17-52 and V-17-53 with water in lieu of air. In Reference 3, however, IEL stated that penetration X-36 would be deleted from the system by capping the penetration on both sides of the containment boundary, and therefore valves V-17-52, V-17-53, and V-17-54 would no longer require testing.

Evaluation

Capping of the penetration on both sides of the containment boundary eliminates these valve from Type C testing requirements since they no longer will be relied upon for any containment isolation function. Consequently, the valves do not require Type C testing and no exemption from Appendix J is required.

3.1.1.5 RCIC and HPCI Condensate Return Isolation Valves (Penetrations X-10 and X-11)

In Reference 3, IEL stated that RCIC condensate return isolation valves CV-2410 and CV-2411 (penetration X-10) and HPCI condensate return isolation valves CV-2211 and CV-2212 (penetration X-11) should be deleted from the list of valves requiring Type C testing because these valves are beyond the second boundary and therefore do not require Type C testing.

Evaluation

The steam side piping of the RCIC and HPCI systems is essentially identical. For simplicity, this evaluation will discuss the RCIC system but will, in effect, apply to both systems.

The RCIC system (steam side) is basically a single-loop system consisting of a 4-inch high pressure steam inlet line, a turbine drive, and a 10-inch condensate return line. The high pressure steam inlet line connects to a 20-inch main steam header inside containment and passes through penetration X-10. Normally open isolation valves MO-2400 and MO-2401 are located in the 4-inch high pressure steam inlet line on both sides of the containment penetration. The condensate return line passes through penetration N-212 and terminates below the water level of the suppression pool. Check valve V-24-23 and locked-open manual globe valve V-24-8 are located in this line, outside of penetration N-212.

A condensate drain pot is located in the high pressure steam line between the outboard isolation valve (MO-2401) and the inlet to the turbine drive. Condensate collected in the drain pot returns to the main condenser via normally open isolation valves CV-2410 and CV-2411. Upon receipt of an RCIC initiation signal, steam line isolation valves MO-2400 and MO-2401 remain open, while condensate return isolation valves CV-2410 and CV-2411 automatically shut to isolate the condensate drain path from the main condenser. Once shut, CV-2410 and CV-2411 cycle intermittently to drain condensate from the drain pot based upon a level control signal operating on drain pot level. At this point, with the RCIC system operating, only valves CV-2410 and CV-2411 prevent leakage of radioactive steam and gases to the atmosphere via the main condenser (in a post-accident condition, there is no guarantee that main condenser off-gas discharge to atmosphere is prevented by the non-safety-related off-gas processing). Once the system is secured or if isolation valves MO-2400 and MO-2401 are shut for other reasons, containment boundary is shifted back to penetrations X-10 and N-212 and leakage past CV-2410 and CV-2411 is no longer significant.

Section II.H of Appendix J requires that containment isolation valves of the main steam system of a boiling water reactor (BWR), as well as containment isolation valves which operate intermittently after an accident, be tested in accordance with Type C testing procedures. Section II.B defines containment isolation valves as those valves which are relied upon to perform a containment isolation function. In view of the foregoing discussion, it is concluded that valves CV-2410 and CV-2411 are relied upon to isolate a potential leakage path from the main steam system of a BWR to the atmosphere during the period when the RCIC system is operating after an accident; therefore, these valves must be Type C tested. Furthermore, a 3/4-inch test line with two isolation valves (V-24-28 and V-24-29) has been located between CV-2410 and CV-2411 specifically to permit this testing. Consequently, IEL's proposal to delete these valves from Type C testing is unacceptable.

Similarly, IEL's proposal to delete HPCI valves CV-2211 and CV-2212 (penetration X-11) from Type C testing is unacceptable. These valves should continue to be Type C tested for the same reasons cited above for the comparable valves in the RCIC system.

3.1.1.6 Main Steam Isolation Valves (Penetrations X-7A, X-7B, X-7C, and X-7D)

In Reference 2, IEL proposed to continue testing main steam line isolation valves (MSIVs) in accordance with existing technical specifications which require testing with air or nitrogen at a pressure of 24 psig between the valves.

Evaluation

Section III.C of Appendix J requires that local leak rate testing be performed at peak calculated accident pressure (Pa), 54 psig at DAEC. Consequently, IEL's proposal requires an exemption from Appendix J to permit the reduced pressure testing.

The main steam system design in most operating BWR plants necessitates leak testing of the MSIVs by pressurizing between the valves. The MSIVs are angled in the main steam lines to afford better sealing in the direction of accident leakage. A test pressure of Pa acting on the inboard disc, however, lifts the disc off its seat; this result in excessive leakage into the reactor vessel and prevents the performance of a meaningful test. Nevertheless, testing by pressurizing between the valves at a reduced pressure is feasible because the reduced pressure does not exert a sufficient force on the disc of the inboard valve to cause it to unseat. It was this consideration which established a valve test pressure of approximately 25 psig during the design stages of the majority of operating BWR units.

From a containment leakage testing standpoint, testing the MSIVs by pressurizing between the valves at a reduced pressure is acceptable because the test results are inherently conservative. In all cases, testing of these valves by exerting a pressure of 54 psig in the direction of accident pressure will result in a larger seating force on the valves than will exist when pressurizing between the valves at reduced pressure. In the case of the inboard valves, testing between the valves is extremely conservative because the test pressure is tending to unseat the inboard valves while accident pressure would always be acting to seat them.

At DAEC, a test pressure of 24 psig was selected because this pressure is equivalent to the column of water against the inboard MSIV when the line between the valve and the reactor vessel is flooded. The significance of this pressure is that it provides the capability to perform the between-the-valves reduced pressure test with zero differential pressure across the inboard MSIV when testing to determine exactly which of the valves may be leaking excessively.

In view of the above discussion, testing of the MSIVs at DAEC by pressurizing between the valves to 24 psig with air or nitrogen is an acceptable exemption to the Type C testing requirements of Appendix J.

3.1.1.7 Valves Water Pressurized Throughout the Accident (Penetrations N-210A & B, N-211A & B, N-224, N-225A & B, N-226, N-227A & B, X-17, X-39A & B)

In Reference 2, IEL listed several valves which it interpreted as not requiring Type C testing in accordance with Appendix J, Section II.H, because these valves were required to remain open or would remain water pressurized for the duration of the accident. In Reference 3, IEL further stated that this containment isolation function was single-active-failure protected, that redundant pumps existed to provide pressurization, that the loops could be cross-connected using cross-ties, and that the loops had redundant valves so that loop pressure could be maintained. The valves in this category were the RHR suppression pool suction, the core spray suppression pool suction, the RCIC and HPCI suppression pool suctions, the LPCI injection, the suppression pool spray, the RHR test line, the vessel head spray, and the containment spray.

Evaluation

Appendix J identifies containment isolation valves which require Type C testing. Section II.B defines containment isolation valves as those valves relied upon to perform a containment isolation function, i.e., those valves which are relied upon in a post-accident condition to prevent the escape of containment air to the outside atmosphere.

The valves which IEL has identified above are part of engineered-safety-feature (ESF) systems and are designed to remain functional after an accident. FRC concurs with IEL that loop pressure can be retained in these systems despite a possible single active failure because of the redundancy designed into the RHR system. The normally shut crosstie valves are not important to this analysis because each RHR loop contains two pumps which are cross-connected by normally open manual valves. However, because of the particular operating characteristics of the RHR system in its LPCI mode, a more detailed review of the specific lines involved is necessary.

The piping configurations of concern are presented in Figures 1 and 2. Figure 1 shows the HPCI, RCIC, and core spray suction lines and one loop of the suction, suppression pool spray, and RHR test lines. Figure 2 shows one loop of LPCI injection, RV head spray, and containment spray. As can be seen in Figure 1, the HPCI, RCIC, core spray, and RHR suction lines are isolated from the containment atmosphere by the water level in the suppression pool. Since these lines are continuously water filled in a post-accident condition,

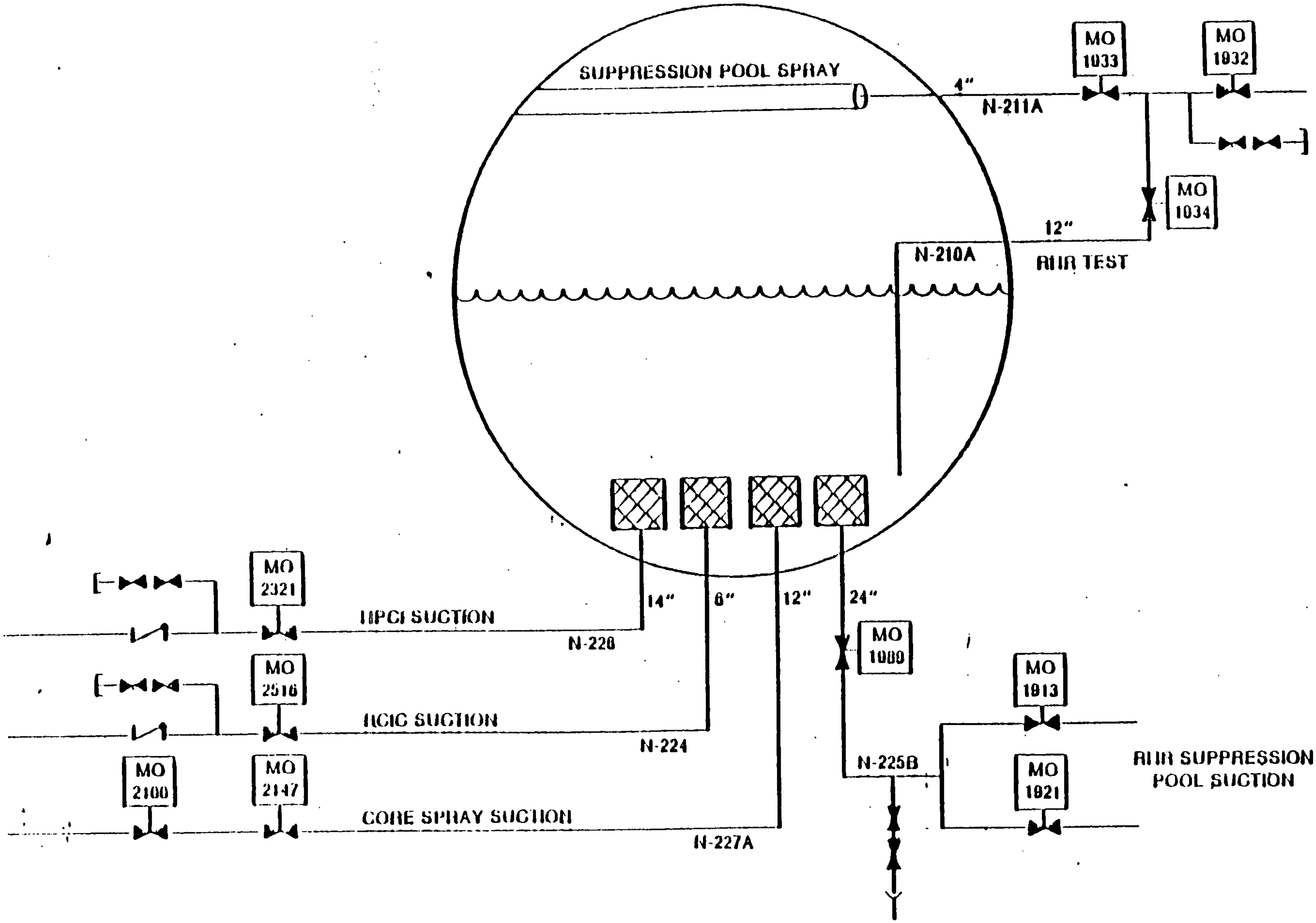
SUPPRESSION POOL SPRAY
4"
N-211A
MO 1033
MO 1032
MO 1034
12"
N-210A
RHR TEST
MO 2321
HPCI SUCTION
N-226
14"
6"
12"
24"
MO 1989
MO 2516
RCIC SUCTION
N-224
MO 1913
N-225B
RHR SUPPRESSION
POOL SUCTION
MO 1921
MO 2100
MO 2147
CORE SPRAY SUCTION
N-227A

Figure 1.

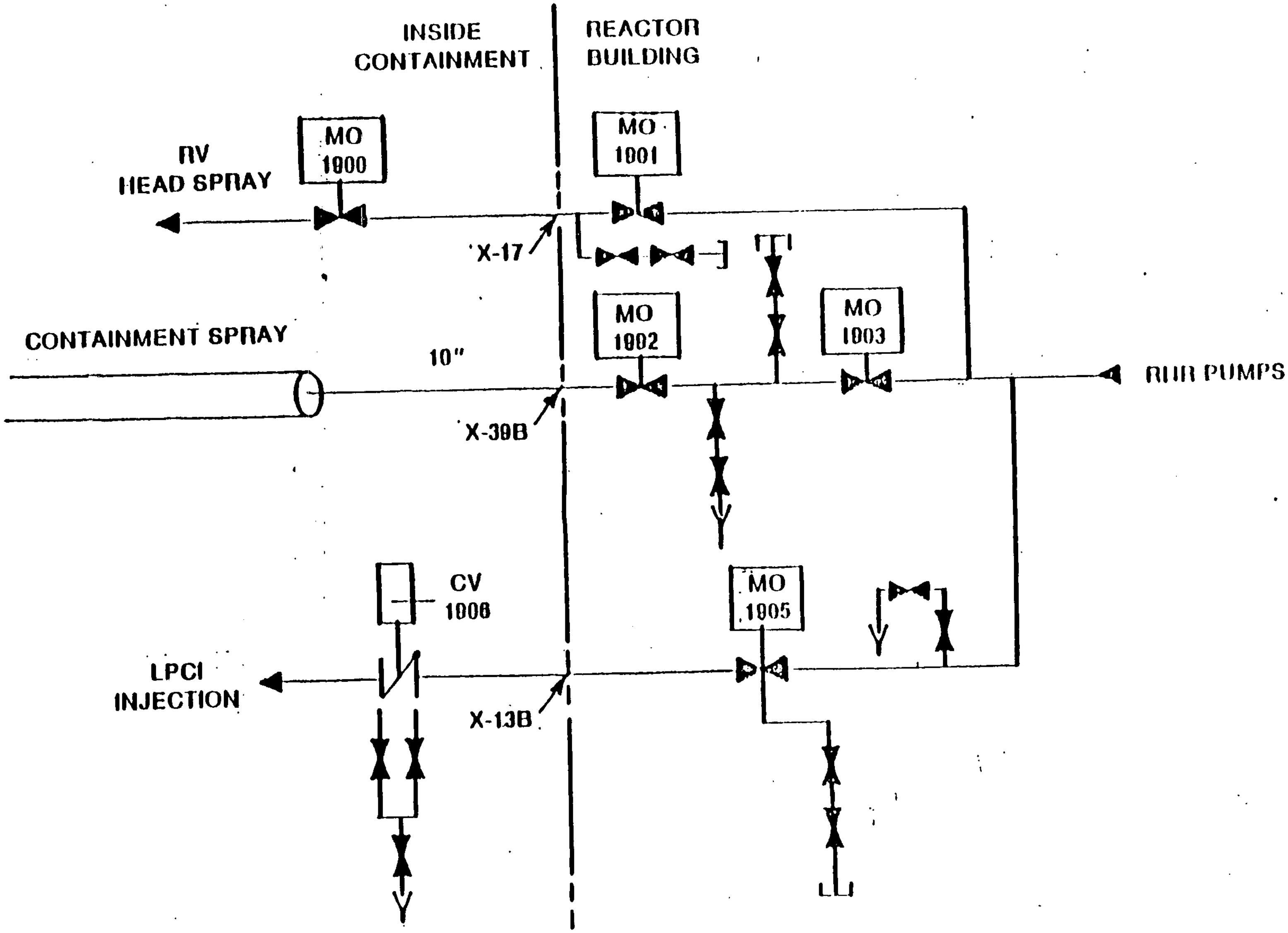
INSIDE CONTAINMENT
REACTOR BUILDING
MO 1900
MO 1001
RV HEAD SPRAY
X-17
CONTAINMENT SPRAY
10"
MO 1002
MO 1003
RHR PUMPS
X-39B
CV 1006
MO 1005
LPCI INJECTION
X-13B

Figure 2.

the isolation valves are not relied upon to prevent the escape of containment air to outside atmosphere; therefore, Type C testing is not required by Appendix J. Similarly, because the RHR test line terminates below the level of the pool, its isolation valve is also isolated from containment atmosphere, and Type C testing of this line is not required.

The LPCI injection line will be normally open and filled with water at a pressure greater than containment accident pressure as soon as safety injection is initiated. Furthermore, should valve MO-1905 (Figure 2) fail to open, the valve will be water sealed by RHR water at pump head pressure, and no single active failure can cause a loss of this pressure. Since MO-1905 is a gate valve, the water pressure will unseat the upstream valve disc and pressurize the valve packing and body-to-bonnet seal area with water. Consequently, there is no path for containment air leakage to the atmosphere through this line, even in the case of air leakage past the seat of check valve CV-1906. Therefore, this line is not a potential source of containment atmosphere leakage and the isolation valves are not required to be Type C tested in accordance with Appendix J.

Unlike the LPCI injection line the remaining three lines (suppression pool spray, containment spray, and RV head spray) are not automatically initiated by safety injection. Flow in these lines is left for manual initiation, if necessary, once sufficient reactor vessel level has been reestablished. Depending upon the severity of the accident, flow in these lines may not be established (particularly containment spray and suppression pool spray). Furthermore, at the start of an accident, there is no guarantee that there is any water in the line between the inboard and outboard isolation valves. In the case of these lines, therefore, there is a potential for containment air to escape to the outside atmosphere through the valve packing or body-to-bonnet seal area of the inboard isolation valve, even though the outboard valve is water sealed, as described in the case of valve MO-1905 of the LPCI injection line.

In the case of the reactor vessel head spray line, the inboard isolation valve is located inside containment (e.g., valve MO-1900). Leakage through the valve packing or body-to-bonnet seal is not a concern since any leakage is merely internal to the containment and does not escape to the outside atmosphere. Consequently, the isolation valves of this line are not relied upon to perform a containment isolation function and do not require Type C testing.

For both the containment spray line and the suppression pool spray lines, however, the inboard isolation valves are located outside containment (e.g., valves MO-1902, MO-1933, MO-1934). If any of these valves leak through the packing or body-to-bonnet seals, the leakage of containment air reaches the outside atmosphere. Consequently, Appendix J requires that these valves be Type C tested. However, since the packing and body-to-bonnet seals are the only potential sources of leakage, the testing may be limited to these particular areas. Valve MO-1902 in the containment spray line is also a gate valve. Testing this valve by pressurizing between valves MO-1902 and MO-1903 achieves the intent of Appendix J because this test will unseat the upstream disc of valve MO-1902 and will pressurize the area of concern. Valves MO-1933 and MO-1934, however, are globe valves. FRC does not have sufficient information to determine whether the packing area is isolated from the containment side of the line when the valve is shut. However, assuming this is the case, these valves may also be tested by pressurizing between valves MO-1932, MO-1933, and MO-1934 since the area of concern will be subjected to the test pressure. If this is not the case, valve MO-1933 must be tested in the direction of accident pressure (note: by pressurizing between the three valves, MO-1934 is tested in the direction of accident pressure since its function in this case is to isolate the suppression pool spray line rather than the RHR test line).

In summary, Type C testing is not required and no exemption is necessary for the following penetrations because Appendix J does not require testing: N-210A & B, N-224, N-225A & B, N-226, N-227A & B, and X-17. For penetration X-39A & B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard isolation valves in order to test the valve packing and body-to-bonnet seals of the inboard valves. For penetration N-211A & B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard valves provided that this testing will expose the packing and body-to-bonnet seal areas of the inboard valves to the test pressure.

3.1.1.8 Submerged Lines (Penetrations N-212, N-214, N-222)

In Reference 2, IEL stated that the suppression pool penetration lines of the RCIC and HPCI turbine exhausts do not require Type C testing since any leakage through these valves would be water leakage because of submergence of the ends of the lines in the suppression pool. In Reference 3, IEL further stated: "Since the leakage will only consist of water, it is considered conservative to add the water leakage to the air leakage and require that the total leakage will remain within the Technical Specification limits."

Evaluation

The valves in question, V-24-8 and V-24-23 (penetration N-212), V-22-16 and V-22-17 (penetration N-214), and V-22-21 and V-22-22 (penetration N-222), are continuously water sealed by the water pressure-head of the suppression pool. The water level of the suppression pool is maintained throughout the post-accident period and therefore any leakage past these valves will be water leakage.

IEL has stated that since any leakage past these valves is water leakage, it is conservative to add the water leakage to the air leakage and to require that the total leakage remain within the technical specification limits. FRC agrees with this statement. Since IEL's proposal is conservative with respect to the requirements of Appendix J, no exemption is required.

3.1.2 Containment Airlocks

In Reference 3, IEL proposed to test containment airlocks at a pressure of Pa and at an interval not longer than one operating cycle. IEL further proposed that whenever the airlock was opened during the operating cycle, and containment integrity was required, the airlock gasket would be tested at Pa following closure if it had been greater than 3 days since the last leakage test.

Evaluation

Appendix J, Section III.D.2 requires that airlocks be tested at 6-month intervals and that airlocks which are opened during the 6-month intervals be tested after each use. Airlocks represent a potentially large leakage path that is more subject to human error than other isolation barriers; therefore, they are tested more often than other isolation barriers. In addition, to ensure that the sealing mechanisms were not damaged during an airlock entry and to ensure that these large potential leakage paths were correctly secured after use, the requirement to test after each use was added.

For certain types of reactors, airlocks have been used frequently. Testing of airlocks after each opening, therefore, may create a situation which results in more rapid degradation of the critical isolation barriers being tested. Moreover, experience obtained since 1969 from the testing of airlocks indicates that only a very few airlock tests have resulted in greater than allowable leakage rates. This infrequent failure of airlock test plus the possibility that excessive testing could lead to a loss of reliability due to equipment degradation leads to the conclusion that testing after each opening may be undesirable. As a compromise between the various interests, the requirement to test after each opening has been defined as within 3 days of each opening or every 3 days during periods of frequent openings. By this definition, the intent of Appendix J that airlock integrity be verified within a reasonable period of time after use is achieved without the excessive testing that would otherwise be required when a series of entries (every few hours) occurs within a short period of time.

IEL's proposal to test airlock gaskets within 3 days of an airlock opening is acceptable. However, IEL's proposal to test the entire airlock at a pressure of Pa once per operating cycle is not acceptable. This proposal does not make adequate allowances to detect potential deterioration of airlocks through normal use, to detect possible damage to the door mechanism, to detect potential damage to door seals through moving equipment into and out of

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containment, and to detect possible fouling of seals during closure. Testing of the entire airlock assembly at a pressure of Pa should be conducted at the 6-month interval required by Appendix J.

3.2 PROPOSED TECHNICAL SPECIFICATION CHANGES

In Reference 3, IEL provided proposed technical specification changes concerning containment leakage rate testing. These changes reflected the proposed exemptions from the requirements of Appendix J discussed in Section 3.1 above as well as other potential changes. IEL stated that all design modifications required to implement the technical specification revisions were anticipated to be completed by the end of the 1980 refueling outage. The following paragraphs provide a technical evaluation of these proposed changes.

3.2.1 Containment Penetrations Subject to Type B Test Requirements (Table 3.7-1)

The proposed revision to Table 3.7-1 provides for changes in the testing requirements for containment airlocks and also adds the requirements to test a flange "O"-ring in penetration 213.

Evaluation

Note 2 of Table 3.7-1 regarding the testing of containment airlocks reads as follows:

> "To be tested at least each operating cycle. Gasket to be tested following closure whenever airlock is opened, providing that containment integrity is required and it has been greater than three (3) days since last leakage test."

As discussed in Section 3.1.2 of this report, the first sentence of this note is unacceptable and should be changed to read: "To be tested at least once every 6 months." The second sentence of the note is acceptable as a requirement of Appendix J as also discussed in Section 3.1.2 of this report.

The addition of the testing requirement for the flange "O"-ring in penetration 213 is in accordance with Appendix J and is therefore acceptable.

Consequently, IEL's proposed revision to Table 3.7-1 is acceptable provided that airlock testing is required at least once every 6 months.

3.2.2 <u>Containment Isolation Valves Subject to Type C Test Requirements (Table 3.7-2)</u>

The proposed revision to Table 3.7-2 provides for the addition and deletion of several valves from this listing of valves which require Type C testing in accordance with Appendix J. Each of the proposed changes to this table is evaluated separately in the following paragraphs.

3.2.2.1 Deletion of Valves Which Do Not Perform a Containment Isolation Function

IEL proposed to delete the following valves from Table 3.7-2 because they do not perform a containment isolation function:

V-14-2	V-14-4	CV-2212
CV-2410	V-17-80	V-17-84
CV-2211	CV-2411	V-22-60

<u>Evaluation</u>

In Section 3.1.1.5 of this report, the deletion of valves CV-2410, CV-2411, CV-2211, and CV-2212 from Type C testing was found unacceptable because, when the RCIC or HPCI systems are in operation after an accident, these valves are relied upon to perform a containment isolation function in view of a potential leakage path from the main steam system of a BWR to the environment. Consequently, these valves should not be deleted from Table 3.7-2.

Valves V-14-2, V-14-4, V-17-80, V-17-84, and V-22-60 do not perform a containment isolation function and can be deleted from Table 3.7-2 since the regulation does not require that they be tested. These valves are normally open manual valves installed to permit testing and/or maintenance of the first containment isolation valve of a particular penetration.

3.2.2.2 Valves Which Do Not Meet the Criteria of Section II.H of Appendix J

IEL proposed to delete valves MO-1908, MO-1909, MO-2115, MO-2117, MO-2135, and MO-2137 from Table 3.7-2 because they do not meet the criteria of Section II.H of Appendix J.

Evaluation

In Section 3.1.1.2 of this report, it was found that valves MO-1908 and MO-1909 do not require Type C testing in accordance with the requirements of Appendix J because they are not relied upon to perform a post-accident containment isolation function. They should be deleted from Table 3.7-2.

In Section 3.1.1.3, however, it was found that valves MO-2115, MO-2117, MO-2135, and MO-2137 should be Type C tested unless the Licensee's testing of the core spray system outside containment is used to demonstrate that the isolation valves remain water sealed throughout the post-accident period. These valves should not be deleted from Table 3.7-2 until such procedures are established.

3.2.2.3 Valves in a Closed System Inside Containment

IEL proposed to delete the following valves from Table 3.7-2 because, in accordance with 10CFR50, Appendix A, GDC 57, the redundant barriers are a single isolation valve outside containment and a closed system inside and, therefore, testing of only the isolation valve outside containment is required:

V-57-62	V-57-65
V-57-66	V-12-65
V-12-64	V-12-63
V-12-62	V-12-66
V-57-61	V-12-68

Evaluation

IEL states that the isolation valves of these penetrations were installed in accordance with GDC 57 and, consequently, only the isolation valve outside containment requires Type C testing. FRC is unable to independently confirm that each of these penetrations qualifies as a GDC 57 penetration under present-day requirements for closed systems. Nevertheless, each of the valves in question is a normally open, manual isolation valve located inside containment. As such, they will be inaccessible under post-accident conditions and are clearly not relied upon to perform a post-accident containment isolation function. Consequently, they are not containment isolation valves in accordance with the definition of Section II.B of Appendix J and therefore do not require Type C testing. FRC concurs with IEL's proposal to delete these valves from Table 3.7-2.

3.2.2.4 Penetration Being Deleted

IEL proposed to delete valves V-17-54, V-17-52, and V-17-53 from Table 3.7-2 because the associated penetration is being deleted.

Evaluation

Based upon IEL's statement in Reference 3 that all modifications necessary to implement the revised technical specifications were anticipated for completion by the end of the 1980 refueling outage, the deletion of these valves from the list of those to be tested is acceptable.

3.2.2.5 Addition of Valves to the Testing List

IEL listed several valves which are to be added to Table 3.7-2. Among others, valves V-24-8, V-24-23, V-22-16, V-22-17, V-22-21, and V-22-22 were added to the table.

Evaluation

With regard to this evaluation, FRC has no comment where the Licensee determines that additional valves should be tested since it only adds conservatism to the containment leakage testing program.

3.2.2.6 Reverse Direction Testing

IEL indicated that certain valves were tested in the direction opposite the pressure existing in a post-accident condition (reverse-direction testing).

In each instance, IEL stated that the results of the reverse-direction testing would be equivalent to or more conservative than testing in the direction of post-accident pressure.

Evaluation

Appendix J, Section III.C, permits reverse-direction testing provided the results are equivalent to or more conservative than results of testing in the direction of post-accident pressure. Consequently, the Licensee's proposed testing is acceptable because it is in accordance with Appendix J. The Licensee should retain onsite documentation of the determination that the reverse-direction testing is equivalent or more conservative than testing in the direction of post-accident pressure.

3.2.3 Miscellaneous Changes to the Technical Specifications

IEL proposed to replace pages 3.7-3 through 3.7-9, 3.7-20 through 3.7-24, 3.7-37, 3.7-38, and 3.7-49 with replacement pages of the same numbers. Table 3-1 of this report provides an evaluation of each of the proposed changes.

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Table 3-1

Proposed Technical Specification Changes

Page No.	IEL's Proposed Wording	Appendix J Requirement	FRC Evaluation
3.7-3	a. Type A Tests		
	7) Initial Leakage Rate Tests		
	a) Prior to initial operation a test shall be performed at 27 psig (Pt, reduced pressure) which is 0.5 Pa, to measure a leakage rate Ltm.	Section III.A.4 requires an initial test be performed at a pressure not less than 0.5 Pa.	The proposed wording complies with Appendix J and therefore is acceptable.
	b) A second test shall be performed at 54 psig (Pa peak pressure) to measure a leakage rate Lam.	Section III.A.4 also requires a second preoperational test be performed at Pa.	The proposed wording complies with Appendix J and therefore is acceptable.
	c) La is defined as the design basis accident leakage rate of 2.0 weight percent of contained air per 24 hours at 54 psig.	Section II.K defines La as the technical specification leakage limit in percent per 24 hours at Pa.	This section complies with Appendix J and therefore is acceptable.
3.7-4	a. Type A Tests		
	8) Periodic Leakage Rate Tests		
	Periodic leakage rate tests shall be performed at peak pressure Pa.	Section III.A.5 permits periodic leak tests to be performed at Pt or Pa.	The proposed wording complies with Appendix J and therefore is acceptable.

Table 3-1 (Cont.)

Page No.	TEL's Proposed Wording	Appendix J Requirement	FRC Evaluation
3.7-4	a. Type A Tests		
	9) Acceptance Criteria		
	Peak pressure test. (Pa), The leakage rate Lam shall be less than 0.75 (La).	Section III.A.5 requires Lam be less than 0.75 La.	The proposed wording complies with Appendix J and therefore is acceptable.
	b. Type B Tests		
	1) Test Pressure		
	All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.	Section III.B.2 requires tests of containment penetrations be performed by local pneumatic pressurization, either individually or in groups, at a pressure not less than Pa.	The proposed wording complies with Appendix J and therefore is acceptable.
3.7-5	c. Type C Tests		
	4) The leakage rate from any containment isolation valve whose seating surface remains water covered post-LOCA, and which is hydrostatically Type C tested, shall be included in the Type C test total. These valves are identified in Table 3.7-2 of this Technical Specification.	Section III.C.2 requires that isolation valves be tested with air or nitrogen as a medium unless sealed by a seal water system.	As discussed in Section 3.1.1.8 of this report, this provision is conservative with respect to the requirements of Appendix J and is therefore acceptable.

Table 3-1 (Cont.)

Page No.	IEL's Proposed Wording	Appendix J Requirement	FRC Evaluation
3.7-6	d. Periodic Retest Schedule		
	2) Type B Tests		
	a) Penetrations and seals of this type (except airlocks) shall be leak tested at 54 psig every other reactor shutdown for major fuel reloading.	Section III.B requires that containment penetrations be tested at a pressure of Pa. For penetrations provided with a pressurization system, Section III.D requires testing at every other shutdown for refueling, not to exceed 3 years (except for airlocks).	The proposed wording should be modified to include the limitation on exceeding 3 years between testings.
	b) The personnel airlock shall be pressurized to 54 psig and leak tested at an interval no longer than one operating cycle. Whenever the airlock is opened during the operating cycle, and containment integrity is required, and it has been greater than (3) days since the last leakage test, the airlock gasket shall be leak tested at 54 psig following airlock closure.	Section III.D.2 requires that containment airlocks be tested at a pressure of Pa once every six months and also after each opening when opened in the interval between 6-month tests.	As discussed in Section 3.1.2 of this report, IEL's proposal to test airlocks once per cycle is unacceptable. This proposed technical specification should be modified to provide for a full airlock test at Pa once every 6 months. IEL's proposal to test airlock gaskets at 54 psig within 3 days of an opening when containment integrity is required is acceptable as discussed in Section 3.1.2 of this report.

Table 3-1 (Cont.)

Page No.	IEL's Proposed Wording	Appendix J Requirement	FRC Evaluation
3.7-8	f. Reporting The Type A test summary report shall include an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. The Type B and C test summary report shall include an analysis and interpretation of the data and the condition of the components which contributed to the failure in meeting the acceptance criteria.	Section V.B.3 requires test results from Type A, B, and C tests that fail to meet acceptance criteria be reported, including an analysis and interpretation of data, the least-squares fit of the data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria.	The proposed wording adequately provides for compliance with the requirements of Appendix J and therefore is acceptable.

4. CONCLUSIONS

FRC has conducted technical evaluations of the outstanding issues pertaining to the implementation of 10CFR50, Appendix J, at DAEC, including the potential requests for exemption from the requirements of Appendix J submitted by IEL in Reference 2 and the proposed changes to the technical specifications at DAEC submitted by IEL in Reference 3. The conclusions resulting from these evaluations are summarized below in the following paragraphs.

Potential Exemptions from Appendix J

- No exemption from Appendix J is required for penetrations X-9A and X-9B as a result of IEL's commitment to modify the inboard feedwater isolation valves.

- Deletion of RHR shutdown cooling supply valves MO-1908 and MO-1909 (penetration X-12) from Type C testing is acceptable because Appendix J does not require testing of these valves. No exemption is required.

- Type C testing of core spray isolation valves MO-2115, MO-2117, MO-2135, and MO-2137 is required unless testing of the core spray system demonstrates that the first isolation valve remains water covered throughout the post-accident period.

- The isolation valves of penetration X-36 (V-17-52, V-17-53, and V-17-54) may be deleted from Type C testing since penetration X-36 will be capped on both sides of the penetration.

- IEL's proposal to delete RCIC and HPCI condensate return isolation valves from Type C testing is unacceptable because the valves are relied upon to perform a containment isolation function (i.e., isolate a direct path to the atmosphere from the main steam system of a BWR) when the RCIC or HPCI systems are in operation after an accident. Valves CV-2410, CV-2411, CV-2211, and CV-2212 should continue to be Type C tested.

- Main steam isolation valves may continue to be tested at 24 psig because the test will provide a conservative measure of the leakage existing at a pressure of Pa due to the design of the valves. Exemption from the Appendix J requirement to test these valves at Pa is acceptable.

- Type C testing is not required and no exemption is necessary for the following penetrations because Appendix J does not require testing: N-210A & B, N-224, N-225A & B, N-226, N-227A & B, and X-17. For penetration X-39B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard isolation valves in order to test the valve packing and body-to-bonnet seals of the inboard valve. For penetration N-211A & B, the inboard isolation valves should be tested in the direction of accident pressure or by pressurizing between the inboard and outboard valves provided that this testing will expose the packing and body-to-bonnet seal areas of the inboard valves to the test pressure.

- IEL's proposal to test the RCIC and HPCI turbine exhaust return lines to the suppression pool (penetrations N-212, N-214, N-222) with water and to add the results to the air leakage totals for compliance with technical specifications limits is acceptable because this proposal is conservative with regard to the requirements of Appendix J.

- A full containment airlock test at a pressure of Pa once every 6 months is required. IEL's proposal to conduct this testing once every operating cycle is unacceptable.

- Testing of airlock gaskets at a pressure of Pa within 3 days of airlock opening is acceptable.

Proposed Technical Specifications Changes

- Note 2 of Table 3.7-1 regarding the testing of containment airlocks should be changed to read "To be tested at least once every 6 months" in lieu of "To be tested at least each operating cycle."

- The addition of a flange "O"-ring to penetration 213 in Table 3.7-1 is acceptable.

- The deletion of valves V-14-2, V-14-4, V-17-80, V-17-84, and V-22-60 from Table 3.7-2 is acceptable because Appendix J does not require that they be tested. Valves CV-2410, CV-2411, CV-2211, and CV-2212 should not be deleted from Table 3.7-2.

- Deletion of valves MO-1908 and MO-1909 from Table 3.7-2 is acceptable because Appendix J does not require that they be tested. Valves MO-2115, MO-2117, MO-2135, and MO-2137 should not be deleted from Table 3.7-2 unless the Licensee's testing of the core spray system is used to demonstrate a water seal on the isolation valves throughout the post-accident period.

- The deletion from Table 3.7-2 of 10 inaccessible, normally open manual valves in closed systems inside containment is acceptable because only the outside valves are relied upon as containment isolation valves in accordance with GDC 57.

- The deletion of V-17-54, V-17-52, and V-17-53 from Table 3.7-2 is acceptable because the associated penetration is being deleted.

- Testing of valves in the direction opposite the pressure existing in the post-accident condition is acceptable because IEL has determined that leakage results are equivalent to or more conservative than leakage results obtained in the direction of post-accident pressure.

- Several miscellaneous changes were found to be acceptable except for the conversion of water leakage to air leakage for certain valves and airlock testing requirements as described above under Potential Exemptions from Appendix J.

5. REFERENCES

1. Mr. Karl Goller, Assistant Director for Operating Reactors
   Letter to Iowa Electric Light and Power Company (IEL)
   August 7, 1975

2. Lee Liu, Vice President, IEL
   Letter to Mr. Karl Goller, Assistant Director for Operating Reactors
   October 13, 1975

3. Lee Liu, IEL
   IEL Application for Amendment of DPR-49 and the Technical Specifications
   to Mr. Harold Denton, Director, Office of Nuclear Reactor Regulation
   August 9, 1978

4. L. O. Root, Assistant Vice President, IEL
   Letter to Mr. T. A. Ippolito, Chief ORB-3
   May 9, 1980

The American Society of Mechanical Engineers

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February 16, 1978

Date 1/8/79
Revision 0

L. T. Harrold, Supervisor, ISI Programs
Washington Public Power Supply System
PO Box 968
3000 George Washington Way
Richland, WA 99352

Subject: Section XI, Division 1, IWA-1100
Scope of Section XI, Division 1

Reference: Your letter of September 19, 1977 (APO 77-59)
ASME File #: BC 77-666
NI 77-371

Dear Mr. Harrold:

Your inquiry and our response are as stated below:

QUESTION:

Is it the intent of Subarticle IWA-1100 that the rules and requirements of Section XI, Division 1 for inservice inspection of Class 1, 2 & 3 pressure retaining components (and their supports) be applied only to water and steam systems in light water cooled nuclear power plants?

REPLY:

Systems containing other than steam or water were not originally considered by the Committee in formulating the rules in Section XI; they may, however, be included for further consideration and for revisions to future editions of Section XI. The requirements shown in Section XI, Article IWA-1000 on Scope and Responsibility, specifically Paragraph IWA-1400, requires the Owner of the nuclear plant to determine the appropriate Code, Class or Classes for each component of the nuclear power plant to be examined according to Section XI rules.

Very truly yours,

Kenneth I. Baron

Kenneth I. Baron,
Assistant Secretary

/fs

P&ID M-143 REVISION 28
SYSTEM : CONTAINMENT ATMOSPHERE CONTROL

IST PROGRAM REVISION : 007 , 11/01/85

VALVE NUMBER	P&ID COOR	IST CLASS	VALVE CAT	VALVE SIZE	VALVE TYPE	ACTUATOR TYPE	NORMAL POSITION	TEST	TEST FREQ	MAXIMUM STROKE TIME	MAXIMUM LEAKAGE	RELIEF REQUEST	REMARKS
CV-4300	C-7	NC	A	18	BTF	AO	C/FC	AT-1	RR		NA	VR-37	
								BTC	OP	005		VR-49	
								FST	OP			VR-17	
								PIT	2Y				
CV-4301	C-8	NC	A	18	BTF	AO	C/FC	AT-1	RR		NA	VR-37	
								BTC	OP	005		VR-49	
								FST	OP			VR-17	
								PIT	2Y				
CV-4302	D-7	NC	A	18	BTF	AO	C/FC	AT-1	RR		NA	VR-37	
								BTC	OP	005		VR-49	
								FST	OP			VR-17	
								PIT	2Y				
CV-4303	D-7	NC	A	18	BTF	AO	C/FC	AT-1	RR		NA	VR-37	
								BTC	OP	005		VR-49	
								FST	OP			VR-17	
								PIT	2Y				
CV-4304	B-7	NC	A	20	BTF	AO	C/FO	AT-1	RR		NA	VR-37	
								BTC	OP	005			
								BTO	OP	005			
								FST	OP			VR-17	
								PIT	2Y				
CV-4305	B-7	NC	A	20	BTF	AO	C/FO	AT-1	RR		NA	VR-37	
								BTC	OP	005			
								BTO	OP	005			
								FST	OP			VR-17	
								PIT	2Y				
CV-4306	E-1	NC	A	18	BTF	AO	C/FC	AT-1	RR		NA	VR-37	
								BTC	OP	005		VR-49	
								FST	OP			VR-17	
								PIT	2Y				

RESPONSE TO QUESTION Q.1

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