Docket

APRIL 20 1981

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Docket No. 50-313

Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

SUBJECT: ORDER FOR MODIFICATION OF LICENSE CONCERNING PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

This letter transmits an Order for Modification of License which revises the Technical Specifications for Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1. The change is a result of the information you provided in response to our 10 CFR 50.54(f) letter of February 23, 1980, regarding primary coolant system pressure isolation valves. Based upon our review of your response, as well as other previously docketed information, we have concluded that a WASH-1400 Event V valve configuration exists at your facility and that corrective action as defined in the attached Order is necessary.

Attached to the Order for Modification of License is the Technical Evaluation Report (TER) which supports the Order; and the plant Technical Specifications which will ensure public health and safety over the operating life of your facility. We are aware that there may be editorial corrections to the attached TER. Please note that the Technical Specifications correctly delineate the requirements for your facility.

In addition to Event V valve configurations, we are continuing our efforts to review other configurations located at high pressure/low pressure system boundaries for their potential risk contribution to an intersystem LOCA. Therefore, further activity regarding the broader topic of intersystem LOCA's may be expected in the future.

CP 3

8104270071

A copy of the enclosed Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

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SURNAME							
DATE							

A copy of the enclosed Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

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

April 20, 1981

Docket No. 50-313

Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

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Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

Arkansas Power & Light Company

cc w/enclosure(s):

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Honorable Ermil Grant
Acting County Judge of Pope County
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Director, Criteria and Standards
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U. S. Environmental Protection Agency
Region VI Office
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Dallas, Texas 75270

Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of ARKANSAS POWER & LIGHT COMPANY (Arkansas Nuclear One, Unit No. 1)	)))))))	Docket No. 50-313

ORDER FOR MODIFICATION OF LICENSE

I

The Arkansas Power & Light Company (the licensee) holds Facility Operating License No. DPR-51, which authorizes the licensee to operate the Arkansas Nuclear One, Unit No. 1 (the facility) at power levels not in excess of 2568 megawatts thermal rated power. The facility, which is located at the licensee's site in Pope County, Arkansas is a pressurized water reactor (PWR) used for the commercial generation of electricity.

II

The Reactor Safety Study (RSS), WASH-1400, identified in a PWR an inter-system loss of coolant accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS contained in-series check valves isolating the high pressure Primary Coolant System (PCS) from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the Event V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.

In order to better define the Event V concern, all light water reactor licensees were requested by letter dated February 23, 1980, to provide the following in accordance with 10 CFR 50.54(f):

1. Describe the valve configurations and indicate if an Event V isolation valve configuration exists within the Class I boundary of the high pressure piping connecting PCS piping to low pressure system piping; e.g., (1) two check valves in series, or (2) two check valves in series with a motor operated valve (MOV);
2. If either of the above Event V configurations exist, indicate whether continuous surveillance or periodic tests are being performed on such valves to ensure integrity. Also indicate whether valves have been known, or found, to lack integrity; and
3. If either of the above Event V configurations exist, indicate whether plant procedures should be revised or if plant modifications should be made to increase reliability.

In addition to the above, licensees were asked to perform individual check valve leak testing prior to plant startup after the next scheduled outage.

By letter dated March 24, 1980, the licensee responded to our February letter. Based upon the review of this response as well as the review of previously docketed information for the facility, I have concluded in consonance with the attached Safety Evaluation (Attachment 1) that one or more valve configuration(s) of concern exist at the facility. The attached Technical Evaluation Report (TER) (Attachment 2) provides, in Section 4.0, a tabulation of the subject valves.

The staff's concern has been exacerbated due not only to the large number of plants which have an Event V configuration(s) but also because of recent unsatisfactory operating experience. Specifically, two plants have leak tested check valves with unsatisfactory results. At Davis-Besse, a pressure isolation check valve in the LPIS failed and the ensuing investigation found that valve internals had become disassembled. At the Sequoyah Nuclear Plant, two Residual Heat Removal (RHR) injection check valves and one RHR recirculation check valve failed because valves jammed open against valve over-travel limiters.

It is, therefore, apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important to safety, they should be tested periodically to ensure low probability of gross failure. As a result, I have determined that periodic examination of check valves must be undertaken by the licensee as provided in Section III below to verify that each valve is seated properly and functioning as a pressure isolation device. Such testing will reduce the overall risk of an inter-system LOCA. The testing mandated by this Order may be accomplished by direct volumetric leakage measurement or by other equivalent means capable of demonstrating that leakage limits are not exceeded in accordance with Section 2.2 of the attached TER.

In view of the operating experiences described above and the potential consequences of check valve failure, I have determined that prompt action is necessary to increase the level of assurance that multiple pressure isolation barriers are in place and will remain intact. Therefore, the public health, safety and interest require that this modification of Facility Operating License No. DPR-51 be immediately effective.

III

Accordingly, pursuant to Section 161i of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT EFFECTIVE IMMEDIATELY, Facility Operating License No. DPR-51 is modified by the addition of the following requirements:

1. Implement Technical Specifications (Attachment 3) which require periodic surveillance over the life of the plant and which specify limiting conditions for operation for PCS pressure isolation valves.
2. If check valves have not been (a) individually tested within 12 months preceding the date of the Order, and (b) found to comply with the leakage rate criteria set forth in the Technical Specifications described in Attachment 3, the MOV in each line shall be closed within 30 days of the effective date of this Order and quarterly Inservice Inspection (ISI) MOV cycling ceased until the check valve tests have been satisfactorily accomplished. (Prior to closing the MOV, procedures shall be implemented and operators trained to assure

that the MOV remains closed. Once closed, the MOV shall be tagged closed to further preclude inadvertent valve opening).

3. The MOV shall not be closed as indicated in paragraph 2 above unless a supporting safety evaluation has been prepared. If the MOV is in an emergency core cooling system (ECCS), the safety evaluation shall include a determination as to whether the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 will continue to be satisfied with the MOV closed. If the MOV is not in an ECCS, the safety evaluation shall include a determination as to whether operation with the MOV closed presents an unreviewed safety question as defined in 10 CFR 50.59(a)(2). If the requirements of 10 CFR 50.46 and Appendix K have not been satisfied, or if an unreviewed safety question exists as defined in 10 CFR 50.59, then the facility shall be shut down within 30 days of the date of this Order and remain shutdown until check valves are satisfactorily tested in accordance with the Technical Specifications set forth in Attachment 3.

4. The records of the check valve tests required by this Order shall be made available for inspection by the NRC's Office of Inspection and Enforcement.

IV

The licensee or any other person who has an interest affected by this Order may request a hearing on this Order within 25 days of its publication in the Federal Register. A request for hearing shall be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy of the request shall also be sent to the Executive Legal Director at the same address, and to Nicholas S. Reynolds, Debevoise & Liberman, 1200 17th Street, Washington, D. C. 20036, attorney for the licensee. If a hearing is requested by a person other than the licensee, that person shall describe, in accordance with 10 CFR 2.714(a)(2), the manner in which his or her interest is affected by this Order. ANY REQUEST FOR A HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

If a hearing is requested by the licensee or other person who has an interest affected by this Order, the Commission will issue an order designating the time and place of any such hearing. If a hearing is held, the issues to be considered at such a hearing shall be:

(a) Whether the licensee should be required to individually leak test check valves in accordance with the Technical Specifications set forth in Attachment 3 to this Order.

(b) Whether the actions required by Paragraphs 2 and 3 of Section III of this Order must be taken if check valves have not been tested within 12 months preceding the date of this Order.

Operation of the facility on terms consistent with this Order is not stayed by the pendency of any proceedings on this Order. In the event that a need for further action becomes apparent, either in the course of proceedings on this Order or any other time, the Director will take appropriate action.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Effective Date: April 20, 1981
Bethesda, Maryland

Attachments:
1. Safety Evaluation Report
2. Technical Evaluation Report
3. Technical Specifications

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

Attachment 1

SAFETY EVALUATION REPORT
ARKANSAS NUCLEAR ONE, UNIT NO. 1
PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES
(WASH-1400, EVENT V)

1.0 Introduction

The Reactor Safety Study (RSS), WASH-1400, identified in a PWR an intersystem loss of coolant accident (LOCA) which is a significant contributor to risk of core melt accidents (Event V). The design examined in the RSS contained in-series check valves isolating the high pressure Primary Coolant System (PCS) from the Low Pressure Injection System (LPIS) piping. The scenario which leads to the Event V accident is initiated by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the LPIS low pressure piping which results in a LOCA that bypasses containment.

In order to better define the Event V concern, all light water reactor licensees were requested by 10 CFR 50.54(f) letter, dated February 23, 1980, to identify valve configurations of concern and prior valve test results, if any. By letter dated March 24, 1980, the licensee responded to our request and this information was subsequently transmitted to our contractor, the Franklin Research Center, for verification that the licensee had correctly identified the subject valve configurations.

2.0 Evaluation

In order to prepare the Technical Evaluation Report (TER) it was necessary that the contractor verify and evaluate the licensee's response to our February 1980 letter. The NRC acceptance criteria used by Franklin were based on WASH-1400 findings, probabilistic analyses and appropriate Standard Review Plan requirements. With respect to the verification of the licensee's response to our information request, the Franklin evaluation was based on FSAR information, ISI/IST site visit data, and other previously docketed information. The attached Franklin TER correctly identifies the subject valve configurations.

3.0 Conclusion

Based on our review of the Franklin TER, we find that the valve configurations of concern have been correctly identified. Since periodic testing of these PCS pressure isolation valves will reduce the probability of an intersystem LOCA we, therefore, conclude that the requirement to test these valves should be incorporated into the plant's Technical Specifications.

Dated: April 20, 1981

8104270 081

THIS REPORT SUPERSEDES ISSUE OF AUGUST 22, 1980

Attachment 2

TECHNICAL EVALUATION REPORT

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS ONE UNIT 1

NRC DOCKET NO. 50-313

NRC TAC NO. 12877

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 210

Prepared by

Franklin Research Center
The Parkway at Twentieth Street
Philadelphia, PA 19103

Author: P. N. Noell
T. C. Stilwell

FRC Group Leader: P. N. Noell

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: P. J. Polk

October 24, 1980

8104270 088

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

1.0 INTRODUCTION

The NRC has determined that certain isolation valve configurations in systems connecting the high-pressure Primary Coolant System (PCS) to lower-pressure systems extending outside containment are potentially significant contributors to an intersystem loss-of-coolant accident (LOCA). Such configurations have been found to represent a significant factor in the risk computed for core melt accidents.

The sequence of events leading to the core melt is initiated by the concurrent failure of two in-series check valves to function as a pressure isolation barrier between the high-pressure PCS and a lower-pressure system extending beyond containment. This failure can cause an overpressurization and rupture of the low-pressure system, resulting in a LOCA that bypasses containment.

The NRC has determined that the probability of failure of these check valves as a pressure isolation barrier can be significantly reduced if the pressure at each valve is continuously monitored, or if each valve is periodically inspected by leakage testing, ultrasonic examination, or radiographic inspection. The NRC has established a program to provide increased assurance that such multiple isolation barriers are in place in all operating Light Water Reactor plants designated by DOR Generic Implementation Activity B-45.

In a generic letter of February 23, 1980, the NRC requested all licensees to identify the following valve configurations which may exist in any of their plant systems communicating with the PCS: 1) two check valves in series or 2) two check valves in series with a motor-operated valve (MOV).

For plants in which valve configurations of concern are found to exist, licensees were further requested to indicate: 1) whether, to ensure integrity of the various pressure isolation check valves, continuous surveillance or periodic testing was currently being conducted, 2) whether any check valves of concern were known to lack integrity, and 3) whether plant procedures should be revised or plant modifications be made to increase reliability.

Franklin Research Center (FRC) was requested by the NRC to provide technical assistance to NRC's B-45 activity by reviewing each licensee's submittal against criteria provided by the NRC and by verifying the licensee's reported findings from plant system drawings. This report documents FRC's technical review.

2.0 CRITERIA

2.1 Identification Criteria

For a piping system to have a valve configuration of concern, the following five items must be fulfilled:

1) The high-pressure system must be connected to the Primary Coolant System;

2) there must be a high-pressure/low-pressure interface present in the line;

3) this same piping must eventually lead outside containment;

4) the line must have one of the valve configurations shown in Figure 1; and

5) the pipe line must have a diameter greater than 1 inch.

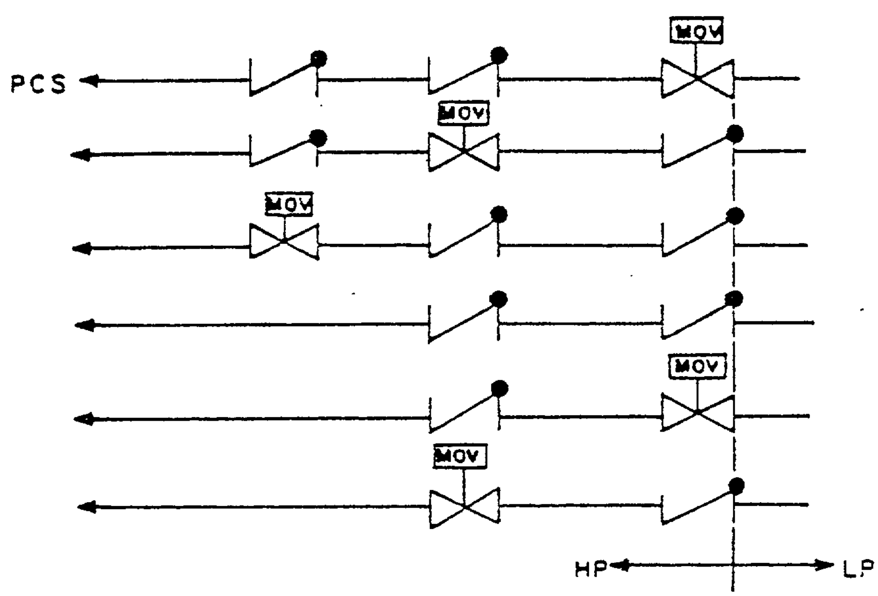

Figure 1. Valve Configurations Designated by the NRC To Be Included in This Technical Evaluation

2.2 Periodic Testing Criteria

For licensees whose plants have valve configurations of concern and choose to institute periodic valve leakage testing, the NRC has established criteria for frequency of testing, test conditions, and acceptable leakage rates. These criteria may be summarized as follows:

2.2.1 Frequency of Testing

Periodic hydrostatic leakage testing* on each check valve shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, each time any check valve may have moved from the fully closed position (i.e., any time the differen- tial pressure across the valve is less than 100 psig), and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

2.2.2 Hydrostatic Pressure Criteria

Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves, having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed leakage shall be adjusted to function maximum pressure differential value. This adjustment shall be made by calculation appropriate to the test media and the ratio between test and function pressure differential, assuming leakage to be directly proportional to the pressure differential to the one-half power.

2.2.3 Acceptable Leakage Rates:

- Leakage rates less than or equal to 1.0 gpm are considered acceptable.
- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount

*To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
- Leakage rates greater than 5.0 gpm are considered unacceptable.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Response to the Generic Letter

In response to NRC's generic letter [Ref. 1] the Arkansas Power and Light Company (APL) stated [Ref. 2] that, "Each low pressure injection system of ANO-1 is protected from the high pressure reactor coolant system by two check valves in series with a motor-operated valve. The configuration is represented schematically in Figure 1. These are the only Event V isolation valve configurations at ANO-1."

The Licensee further stated "The integrity of DH-14A & B [check valves] in conjuction with DH-13A & B [checks valves] is assured by monitoring total RCS leakage. The integrity of DH-14A & B is further assured, in conjunction with CF-1A & B [checks valves], by monitoring core flood tank level and pressure. None of these have been known, or found, to lack integrity."

It is FRC's understanding that, with APL's concurrence, NRC will direct APL to change its Plant Technical Specifications as necessary to ensure that periodic leakage testing (or equivalent testing) is conducted in accordance with the criteria of Section 2.2.

3.2 FRC Review of Licensee's Response

FRC has reviewed the licensee's response against the plant-specific Piping and Instrumentation Diagrams (P&IDs) [Ref. 3] that might have the valve configurations of concern.

FRC has also reviewed the efficacy of instituting periodic testing for the check valves involved in this particular application with respect to the reduction of the probability of an intersystem LOCA in the Decay Heat Removal piping lines.

In its review of the P&IDs [Ref. 3] for the Arkansas One Unit 1, FRC found the following piping system to be of concern:

The Decay Heat Removal System (DHR) is composed of two piping trains (A and B) each connected directly to the reactor vessel. Each train has two check valves and a motor-operated valve in one of the series configurations of concern. In each train the high-pressure/low-pressure interface is located on the upstream side of the motor-operated valve (MOV). These valves are listed below:

Decay Heat Removal System

Train A

high-pressure check valve, DH14A
high-pressure check valve, DH13A
high-pressure MOV, CV-1401, normally closed

Train B

high-pressure check valve, DH14B
high-pressure check valve, DH13B
high-pressure MOV, CV-1400, normally closed

In accordance with the criteria of Section 2.0, FRC has found no other valve configurations of concern existing in this plant. These findings confirm the licensee's response [Ref. 2].

FRC reviewed the effectiveness of instituting periodic leakage testing of the check valves in these lines as a means of reducing the probability of an intersystem LOCA occurring. FRC found that introducing a program of check valve leakage testing in accordance with the criteria summarized in Section 2.0 will be an effective measure in substantially reducing the probability of an intersystem LOCA occurring in these lines, and a means of increasing the probability that these lines will be able to perform their safety-related functions. It is also a step toward achieving a corresponding reduction in the plant probability of intersystem LOCA in the Arkansas One Unit 1.

4.0 CONCLUSION

Arkansas One Unit 1 has been determined to have valves in one of the configurations of concern in both A and B trains of the Decay Heat Removal System.

If APL modifies the Plant Technical Specification for Arkansas One Unit 1 to incorporate periodic testing, as delineated in Section 2.2, for the check valves itemized in Table 1.0, then FRC considers this an acceptable means of achieving plant compliance with the NRC staff objectives of Reference 1.

Table 1.0

Primary Coolant System Pressure Isolation Valves

System	Check Valve No.	Allowable Leakage*
Decay Heat Removal		
Train A	DH14A DH13A	
Train B	DH14B DH13B	

5.0 REFERENCES

[1]. Generic NRC letter, dated 2/23/80, from Mr. D. G. Eisenhut, Department of Operating Reactors (DOR), to Mr. C. L. Steel, Arkansas Power and Light Company (APL).

[2]. Arkansas Power and Light Company's response to NRC's letter, dated 3/24/80, from Mr. C. L. Steel (APL) to Mr. D. G. Eisenhut (DOR).

[3]. List of examined P&IDs:

Arkansas Power and Light drawings:

Fig. 9-1

Fig. 9-3

*To be provided by licensee at a future date in accordance with Section 2.2.3.

Fig. 9-4

Fig. 9-5

Fig. 9-12

Bechtel Drawings:

M-200, (Rev. 4)

M-201, (Rev. 2)

M-230, (Rev. 16)

M-231, (Rev. 15)

M-232, (Rev. 13)

M-233, (Rev. 12)

M-234, (Rev. 15) Sh. 1 of 2

M-234, (Rev. 11) Sh. 2 of 2

M-236, (Rev. 12)

Attachment 3

ATTACHMENT TO ORDER FOR MODIFICATION OF

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages contain vertical lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

28

29a (new)

73

73a

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc. except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3, the rate of cooldown and the conditions of shutdown shall be determined by the safety evaluation for each case and reported as required by Specification 6.12.3.

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.6 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.7 When the reactor is at power operation, 3 reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided 2 other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1 and 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

3.1.6.9 If the reactor coolant system pressure isolation valve leakage is greater than the values given in Table 3.1.6.9, isolate (by having at least two valves in the high pressure piping closed*) the high pressure portion of the affected system from the low pressure portion within 4 hours and apply Specification 3.3.6, or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage, including evaporative losses (which may be on the order of 0.5 gpm), to prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, the first indication of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also available even during a loss of off-site power.

If leakage is to the reactor building it may be identified by one or more of the following methods:

a. Leakage is monitored by a level indicator in the reactor building sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as the reactor coolant system, service water system, intermediate cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The reactor building sump contains 63.6 gallons per inch of height. A 1 gpm leak would be detected in less than 1 hour.

*The motor operated valve shall remain closed and power supplies deenergized.

TABLE 3.1.6.9

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[a][b][c] Allowable Leakage
Decay Heat Removal Train A	DH-14A	≤ 5.0 GPM
	DH-13A, DH-17	≤ 5.0 GPM (both valves together total)
Decay Heat Removal Train B	DH-14B	≤ 5.0 GPM
	DH-13B, DH-18	≤ 5.0 GPM (both valves together total)

Footnote:

[a] 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

[b] Minimum differential test pressure shall not be less than 150 psig.

[c] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Table 4.1-2

Minimum Equipment Test Frequency

	Item	Test	Frequency
1.	Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown
2.	Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3.	Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Months
4.	Main Steam Safety Valves	Setpoint	Four Valves Every 18 Months
5.	Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6a.	Reactor Coolant System Leakage	Evaluate	Daily
b.	Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2
7.	Emergency-powered Pressurizer Heaters	Power availability	Daily
		Heater capacity functional test	Every 18 Months
8.	Reactor Building Isolation Trip	Functioning	Every 18 months
9.	Service Water Systems	Functioning	Every 18 months
10.	Spent Fuel Cooling System	Functioning	Every 18 months when irradiated fuel is in the pool

1/ Same as tests listed in Section 4.7

Notes:

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

Item	Test	Frequency
11. Decay Heat Removal System Isolation Valve Automatic Closure and Isolation System	Functioning	Every 18 months
12. Flow Limiting Annulus on Main Feedwater Line at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.
13. SLBIC Pressure Sensors	Calibrate	Every 18 Months.
14. Main Steam Isolation Valves	a. Excercise Through Approximately 10% Travel	a. Quarterly
	b. Cycle	b. Every 18 Months.
15. Main Feedwater Isolation Valves	a. Exercise Through Approximately 5% Travel	a. Quarterly
	b. Cycle	b. Every 18 Months.
16. Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs (applied vertically upward).	Each refueling shutdown.