Docket

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President
Jersey Central Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

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 Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. 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 contrary to our letter dated August 12, 1980, 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.

OFFICE							
SURNAME							
DATE							

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

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

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

April 20, 1981

Docket No. 50-219
LS05-81-04-024

Mr. I. R. Finfrock, Jr.
Vice President
Jersey Central Power & Light Company
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Finfrock:

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Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

cc
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Forked River, New Jersey 08731

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of JERSEY CENTRAL POWER & LIGHT CO. (Oyster Creek Nuclear Generating Station)	))))))	Docket No. 50-219

ORDER FOR MODIFICATION OF LICENSE

I

The Jersey Central Power & Light Company (the licensees) hold Provisional Operating License No. DPR-16, which authorizes the licensees to operate the Oyster Creek Nuclear Generating Station (the facility) at power levels not in excess of 1930 megawatts(thermal) rated power. The facility, which is located at the licensee's site in Ocean County, New Jersey is a Boiling Water Reactor (BWR) 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 17, 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 Provisional Operating License No. DPR-16 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, Provisional Operating License No. DPR-16 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 G. F. Trowbridge, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Stree, N. W., 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 preceeding 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

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

Attachment 1

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

SAFETY EVALUATION REPORT
OYSTER CREEK NUCLEAR GENERATING STATION
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 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 17, 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 attached 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

Attachment 2

THIS REPORT SUPERSEDES ISSUE OF JULY 21, 1980

TECHNICAL EVALUATION REPORT

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

JERSEY CENTRAL POWER AND LIGHT COMPANY
OYSTER CREEK UNIT 1

NRC DOCKET NO. 50-219

NRC TAC NO. 12919

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 252

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

March 20, 1980

8105040041

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.

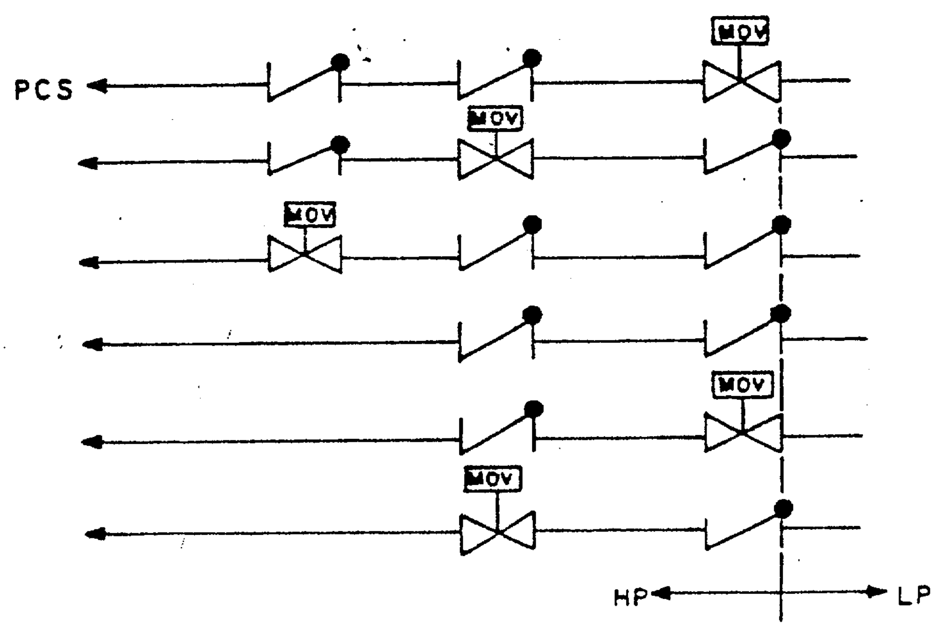

Figure 1. Valve Configurations Designated by the NRC To Be Included in This Technical Evaluation

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* (ultrasonic testing or radiographic 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 differential 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 (as low as reasonably achievable) requirements, pressure indicators may be used in accordance with approved procedures as documented by calculation to determine the need for hydrostatic testing.

that reduces the margin between the 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 the NRC's generic letter [Ref. 1], the Jersey Central Power and Light Company (JCP) stated [Ref. 2] that, "Based on the review of the systems at Oyster Creek, it has been concluded that the valve configurations at Oyster Creeks are not as described in your letter."

The licensee did, however, include a description of the valve configurations of concern existing in the Core Spray System at Oyster Creek Unit 1.

It is FRC's understanding that, with JCP's concurrence, the NRC will direct JCP 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 Core Spray System pipe lines.

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

- The valve configuration of concern, existing in both the A and B loops of the Core Spray System, consists initially of two parallel air-operated check valves inside the Drywell, leading away from the Reactor Vessel. These two check valves join together in a common line leading to two parallel, normally closed, motor-operated valves (MOVs) outside the containment. These two MOVs then join in a common line to a single, locked open MOV. The high-pressure/low-pressure interface was indicated by the Licensee to exist at the upstream side of this single, locked open, MOV. The appropriate valves of the Core Spray System are listed below:

Core Spray System

Loop A

high-pressure, air-operated, parallel check valve, NZ02A

high-pressure, air-operated, parallel check valve, NZ02C

high-pressure, parallel MOV, V-20-15, normally closed (n.c.)

high-pressure, parallel MOV, V-20-40, n.c.

high-pressure, MOV, V-20-12, locked open (l.o.)

Loop B

high-pressure, air-operated, parallel check valve, NZ02B

high-pressure, air-operated, parallel check valve, NZ02D

high-pressure, parallel MOV, V-20-21, n.c.

high-pressure, parallel MOV, V-20-41, n.c.

high-pressure, MOV, V-20-18, l.o.

In accordance with the criteria of Section 2.0, FRC found no other valve configurations of concern existing in this plant.

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 an intersystem LOCA in Oyster Creek Unit 1.

4.0 CONCLUSION

Based on the previously docketed information and drawings made available for FRC review, FRC found that loops A and B of the Core Spray System in Oyster Creek Unit 1 contain a valve configuration of concern (identified in Figure 1). Thus, if the licensee's review of the valving configuration contained in the loops A and B of the Core Spray System confirms FRC's finding,then the valve configurations of concern existing in Oyster Creek Unit 1 incorporate the valves listed in Table 1.0.

If JCP modifies the Plant Technical Specifications for Oyster Creek 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*
Core Spray		
Loop A	NZ02A NZ02C	
Loop B	NZ02B NZ02D	

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

5.0 REFERENCES

1. Generic NRC letter, dated 2/23/80, from Mr. D. G. Eisenhut, Department of Operating Reactors (DOR), to Mr. I. R. Finbrock, Jr., Jersey Central Power and Light Company (JCP).

2. Jersey Central Power and Light Company's response to the generic NRC letter, dated 3/17/80, from Mr. I. R. Finbrock (JCP) to Mr. D. G. Eisenhut (DOR).

3. List of examined P&IDs:

   General Electric Drawings of Oyster Creek Unit 1:

148P711	(Rev. 7)
148F444	(Rev. 11)
148F723	(Rev. 5)
197E871	(Rev. 7)
237E487	(Rev. 15)
237E726	(Rev. 10)
237E798	(Rev. 10)
2406	(Rev. 8)
706E249	(Rev. 3)
885D781	(Rev. 11)
886D403	(Rev. 2)

ORDER FOR MODIFICATION OF LICENSE

(EVENT V)

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

Insert the following pages in the Appendix A Technical Specifications

Pages

3.3-2*

3.3-2a

3.3-2b*

3.3-8

4.3-1*

4.3-1a

4.3-2*

4.3-9

*There are no changes to the provisions contained on this page; it is merely included for pagination purposes.

D. Reactor Coolant System Leakage

Reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total leakage in the containment, identified and unidentified, shall not exceed 25 gpm. If these conditions cannot be met, the reactor will be placed in the cold shutdown condition.

E. Reactor Coolant Quality

1. The reactor coolant quality shall not exceed the following limits during power operation with steaming rates to the turbine-condenser of less than 100,000 pounds per hour.

conductivity	2 μ mho/cm
chloride ion	0.1 ppm

2. The reactor coolant quality shall not exceed the following limits during power operation with steaming rates to the turbine-condenser of at least 100,000 pounds per hour.

conductivity	10 mho/cm
chloride ion	1.0 ppm

3. If Specification 3.3.E.1 and 3.3.E.2 cannot be met, the reactor shall be placed in the cold shutdown condition.

F. Recirculation Loop Operability

1. The reactor shall not be operated with one or more recirculation loops out of service except as specified in Specification 3.3.F.2.

2. Reactor operation with one idle recirculation loop is permitted provided that the idle loop is not isolated from the reactor vessel.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met the reactor shall be placed in the cold shutdown condition within 24 hours.

G. Primary Coolant System Pressure Isolation Valves

Applicability:

Operational Conditions - Startup and Run Modes; applies to the operational status of the primary coolant system pressure isolation valves.

Objective:

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification:

1. During reactor power operating conditions, the integrity of all pressure isolation valves listed in Table 3.3.1 shall be demonstrated. Valve leakage shall not exceed the amounts indicated.

2. If Specification 1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Bases: The reactor coolant system[1] is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The Oyster Creek reactor vessel was designed and manufactured in accordance with General Electric Specification 21A1105 and ASME Section I as discussed in Reference 13. The original operating limitations were based upon the requirement that the minimum temperature for pressurization be at least 60°F greater than the nil ductility transformation temperature. The minimum temperature for pressurization at any time in life had to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Figures 3.3.1 is derived from an evaluation of the fracture toughness properties performed for Oyster Creek. (Reference 12) in an effort to establish new operating limits. The results of neutron flux dosimeter analyses in Reference 12 indicate that the total fast neutron fluence (>1 Mev) expected for Oyster Creek at the end of ten effective full power years of operation is 1.22×10^{18} nvt on the inside surface of the reactor vessel core region shell. A conservative fast neutron fluence of 75% of this value is assumed at the 1/4 T (one quarter of wall thickness) location for the preparation of the pressure/temperature curves in Figure 3.3.1.

Stud tensioning is considered significant from the standpoint of brittle fracture only when the preload exceed approximately 1/3 of the final design value. No vessel or closure stud minimum temperature requirements are considered necessary for preload values below 1/3 of the design preload with the vessel depressurized since preloads below 1/3 of the design preload result in vessel closure and average bolt stresses which are less than 20% of the yield strengths of the vessel and bolting materials. Extensive service experience with these materials has confirmed that the probability of brittle fracture is extremely remote at these low stress levels, irrespective of the metal temperature.

TABLE 3.3.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum Allowable Leakage[a]
Core Spray System 1	NZ02A	5.0 GPM
	NZ02C	5.0 GPM
Core Spray System 2	NZ02B	5.0 GPM
	NZ02D	5.0 GPM

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.

5. Test differential pressure shall not be less than 150 psid.

4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Neutron flux monitors shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitors shall be removed and tested at the first refueling outage to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT from Figure 3.3.1.

B. Non-destructive examinations shall be made on the components as specified in Table 4.3.1. Any indication of a defect shall be investigated and evaluated.

C. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system. The requirements of specification 3.3.A shall be met during the test.

D. Each replacement safety valve or valve that has been repaired shall be bench checked for the proper set point. A minimum of 5 of the valves shall be bench checked or replaced with a bench checked valve each refueling outage such that all valves are checked in three successive refueling outages, to insure set points are as follows:

Number of Valves	Set Point (psig)
4	1212 ± 12
4	1221 ± 12
4	1230 ± 12
4	1239 ± 12

E. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

F. Primary Coolant System Pressure Isolation Valves

Specification:

1. Periodic leakage testing[a] on each valve listed in table 4.3.2 shall be accomplished prior to exceeding 600 psig reactor pressure every time the plant is placed on 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 preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

[a] 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.

Basis:

Numerous data are available relating integrated flux and the change in Nil-Ductility Transition Temperature (NDTT) in various steels. The base metal has been demonstrated to be relatively insensitive to neutron irradiation (see expected NDT changes in FDSAR Table IV-1-1, and Figures IV-2-9 and IV-2-10). The most conservative data has been used in

Specification 3.3. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and if necessary correct, the calculated data to determine an accurate flux. From this a conservative NDT temperature can be determined. Since no shift will occur until an integrated flux of 10^{17} nvt is reached, the confirmation can be made long before an NDTT shift would occur.

Prior to operation the reactor coolant system will be free of gross defects and the facility has been designed such that gross defects should not occur throughout life; however, to determine the status of the coolant system to ensure that gross defects are not developing this surveillance program was developed. This inspection will reveal problem areas should they occur before a leak develops. In addition, extensive visual inspection for leaks will be made on critical systems. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the AEC. These studies show that it requires thousands of stress cycles, at stresses beyond any conceived in a reactor system to propagate a crack and it is thus concluded that the frequency is adequate. The access provisions for in-service inspection has been compared with the access requirements of the proposed N-45 Code for In-Service Inspection of Nuclear Reactor Coolant Systems. The degree of access required by N-45 is not generally available, however, volumetric inspection of accessable areas has been proposed. It is considered appropriate to evaluate the results obtained from compliance with this Technical Specification and the state of the art before establishing a long term inspection program.

Experience in safety valve operation shows that a check of approximately 1/3 of the safety valves per year is adequate to detect failures or deterioration. The tolerance value is specified in Section I of the ASME Code at ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 12 psig higher the safety limit of 1375 psig is not exceeded.

TABLE 4.3.2

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a)] Allowable Leakage
Core Spray System 1	NZ02A	5.0 GPM
	NZ02C	5.0 GPM
Core Spray System 2	NZ02B	5.0 GPM
	NZ02D	5.0 GPM

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.

5. Test differential pressure shall not be less than 150 psid.