Docket File

APR 20 1981

Docket No. 50-~~346~~ 305

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Eugene R. Mathews, Vice President
Power Supply and Engineering
Wisconsin Public Service Corporation
Post Office Box 1200
Green Bay, Wisconsin 54305

Dear Mr. Mathews:

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-43 for the Kewaunee Nuclear Power Plant. 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.

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OFFICE							
SURNAME							
DATE							

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

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
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Docket

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

April 20, 1981

Docket No. 50-~~346~~ 305

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Power Supply and Engineering
Wisconsin Public Service Corporation
Post Office Box 1200
Green Bay, Wisconsin 54305

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Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Order for Modification
of License

cc w/enclosure:
See next page

Mr. Eugene R. Mathews
Wisconsin Public Service Corporation

cc: Steven E. Keane, Esquire
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Milwaukee, Wisconsin 53202

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822 Juneau Street
Kewaunee, Wisconsin 54216

Stanley LaCrosse, Chairman
Town of Carlton
Route 1
Kewaunee, Wisconsin 54216

Mr. Donald L. Quistroff, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

Chairman
Public Service Commission of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Mr. Patrick Walsh
Assistant Attorney General
114 East, State Capitol
Madison, Wisconsin 53702

U. S. Nuclear Regulatory Commission
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Kewaunee, Wisconsin 54216

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
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of Wisconsin Public Service Corporation Et Al. (Kewaunee Nuclear Power Plant)	)	Docket No. 50-305

ORDER FOR MODIFICATION OF LICENSE

I

The Wisconsin Public Service Corporation Et al.(the licensee) holds Facility Operating License No. DPR-43, which authorizes the licensee to operate the Kewaunee Nuclear Power Plant at power levels not in excess of 1650 megawatts thermal rated power. The license was originally issued on December 21, 1973 and will expire on Midnight, August 6, 2008. The facility, which is located at the licensee's site in Kewaunee County, Wisconsin, 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 18, 1980 the licensee responded to our February letter. Based upon the NRC 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-43 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-43 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 Steven E. Keane, Esquire, Faley and Lardner, 777 East Wisconsin Avenue, Milwaukee, Wisconsin, 53202, 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
KEWAUNEE NUCLEAR POWER PLANT
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 18, 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

ATTACHMENT 2

TECHNICAL EVALUATION REPORT

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

WISCONSIN PUBLIC SERVICE CORPORATION KEWAUNEE UNIT 1

NRC DOCKET NO. 50-305

NRC TAC NO. 12928

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC TASK 259

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

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.

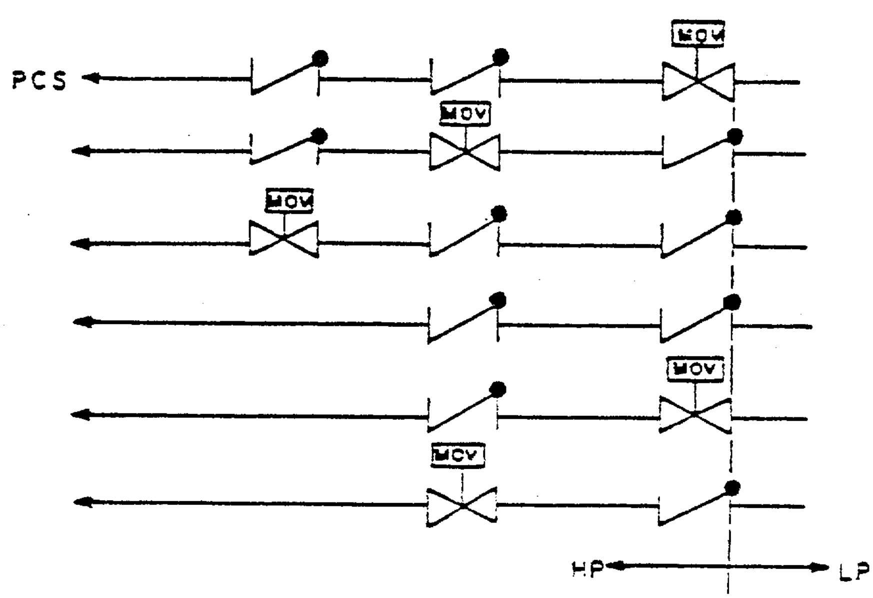

Figure 1. Valve Configurations Designated by the NRC To Be Included in This Technical Evaluation

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 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 Wisconsin Public Service Corporation (WPS) stated [Ref. 2] that, "We have reviewed the Kewaunee Plant design and have identified one system that utilizes the design noted above. The Kewaunee Nuclear Power Plant utilizes the residual heat removal (RHR) pumps for low-pressure safety injection into the upper plenum of the reactor vessel. There are two low head safety injection lines penetrating the upper plenum; each line has two check valves and one motor operated valve which serve to isolate the low pressure piping from the reactor coolant system."

The licensee further stated, "The operator has indication of RHR system pressure in the control room from the RHR pump discharge pressure. Should the RHR loop pressurize due to leakage past the isolation check valves, an RHR pump discharge high pressure alarm would occur, alerting the operator of the condition. This, in effect, provides continuous surveillance on the isolation check valves."

It was discovered by FRC that a crossover piping line between the Loop B cold leg accumulator and the Residual Heat Removal System contains also a valve configuration of concern.

It is FRC's understanding that, with WPS's concurrence, the NRC will direct WPS 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 Residual Heat Removal and the Loop B cold leg accumulator/RHR crossover piping lines.

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

> The Residual Heat Removal System, used for Low-Head Safety Injection, is connected directly to the Reactor Vessel via two separate piping branches A and B. Each branch has a valve configuration of concern consisting of two check valves in series with a motor-operated valve (MOV). The high-pressure/low-pressure interface exists at the upstream side of the MOV.
>
> The crossover line between the Loop B cold leg accumulator and the Residual Heat Removal System lines contains a configuration of concern consisting of a single check valve in series with a MOV. The high-pressure/low-pressure interface exists also at the upstream side of the MOV. The appropriate valves for both piping systems are listed below:

Residual Heat Removal/Low-Head Safety Injection System

<u>Reactor Vessel</u>

Branch A

- high-pressure check valve, SI-304A
- high-pressure check valve, SI-303A
- high-pressure MOV, SI-302A, normally open (n.o.)

Branch B

- high-pressure check valve, SI-304B
- high-pressure check valve, SI-303B
- high-pressure MOV, SI-302B, n.o.

Loop B, Cold Leg, Accumulator to RHR Crossover Line

- high-pressure check valve, SI-22B
- high-pressure MOV, RHR11

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 Kewaunee Unit 1.

4.0 CONCLUSION

It has been determined that the Residual Heat Removal/Low-Head Safety Injection system in Kewaunee Unit 1, incorporates valving in one of the configurations (identified in Figure 1) designated by the NRC as a valve configuration of concern. Moreover, based on the previously docketed information and drawings made available for FRC review, FRC found that the crossover line between the Loop B cold-leg Accumulator and the Residual Heat Removal/Low-Head Safety Injection Systems lines also incorporates a valve configuration of concern. Thus, if the licensee's review of the valving configuration contained in this crossover line confirms FRC's finding, then valve configurations of concern exist in two systems of Kewaunee Unit 1 and incorporate the valves listed in Table 1.0.

If WPS modifies the Plant Technical Specifications for Kewaunee 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*
Residual Heat Removal/ Low-Pressure Safety Injection		
Reactor Vessel		
Branch A	SI-304A SI-303A	
Branch B	SI-304B SI-303B	
Loop B, Cold Leg Accumulator to RHR Crossover Line		
	SI-22B	

*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. E. R. Mathews, Wisconsin Public Service Corporation (WPS).

2. Wisconsin Public Service Corporation's response to NRC's letter, dated 3/18/80, from Mr. E. R. Mathews (WPS) to Mr. D. G. Eisenhut (DOR).

3. List of examined P&IDs:

    FSAR Drawings of Kewaunee Unit 1:

    Fig. 6.2-1

    Fig. 6.2-2

    Fig. 9.3-1

    Fig. 9.3-2

    Fig. 9.3-3

    Fig. 9.4-1

ATTACHMENT 3

TECHNICAL SPECIFICATIONS CHANGES

Remove pages	Insert pages
TS 3.1-1	TS 3.1-1
TS 3.1-2	TS 3.1-2
-------	TS 3.1-2a
Table TS 3.1-2	Table TS 3.1-2
TS 4.2-2.A	TS 4.2-2.A
TS 4.2-8	TS 4.2-8
TS 4.2-8A	TS 4.2-8A

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3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specifications

a. OPERATIONAL COMPONENTS

Specification:

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the operating mode of operation, except for low power tests, both reactor coolant pumps shall be in operation.

2. Steam Generator

   A. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F.

   B. Reactor power shall not be maintained above 10% of rated power when one steam generator is isolated.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power operated relief valves are set for test pressure plus 35 psi and the charging pump has a safety valve to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

Applicability:

Operational defined as Operating, and Hot Standby.

Objective

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

Specification:

A. All pressure isolation valves listed in Table TS 3.1-2 shall be functional as a pressure isolation device, except as specified in B. Valve leakage shall not exceed the amounts indicated.

B. In the event that integrity of any pressure isolation valve as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition. (a)

C. If Specification A and B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Hot Shutdown condition within the next 4 hours, the Intermediate Shutdown condition in the next 6 hours and the Cold Shutdown condition within the next 24 hours.

(a) Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

Part 1 of the specification requires that both reactor coolant pumps be operating when the reactor is in power operation to provide core cooling in the event that a loss of flow occurs. Planned power operation with one loop out of service is not allowed in the present design because the system does not meet the single failure (locked rotor) criteria requirement for this mode of operation. The flow provided in each case in Part 1 will keep DNBR well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. One pump operation is not permitted for any length of time except for tests. Upon loss of one pump below 10% full power the core power shall be reduced to a level below the maximum power determined for zero power testing. Natural circulation will remove decay heat up to 10% power. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. (1)

Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at set point. Below 350°F and 350 psig, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate protection against over-pressurization.

The Basis for the Pressure Isolation Valves is contained with Reference 2.

References:

(1) FSAR Section 7.2.2

(2) Order for Modification of License dated

TABLE T.S. 3.1-2

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a) (b)] Allowable Leakage
Reactor Vessel, Core Flooding Line (Upper Plenum Injection)		
SI - 304A		≤5.0 Gallons per Minute
SI - 303A		≤5.0 Gallons per Minute
SI - 304B		≤5.0 Gallons per Minute
SI - 303B		≤5.0 Gallons per Minute
Loop B 12" Accumulator Discharge Line		
SI - 22B		≤5.0 Gallons per Minute

FOOTNOTES:

(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 test differential pressure shall not be less than 150 psid.

10. The Following Surveillance Tests Be Undertaken:

a. Periodic leakage testing (1) on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the operating mode after every time the plant is placed in the cold shutdown condition for refueling, affer 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, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

(1) 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.

The exclusion criteria of IS-121 have been applied to determine which parts of systems or components are subject to surface or volumetric examinations and which parts are subject to a visual examination for evidence of leakage during the system hydrostatic test. A description of the system boundaries, delineating those parts subject to volumetric examination, those parts subject to surface examination and those parts requiring visual inspection during hydro are given in the notes to FSAR Table 4.4-2, titled Tables 4.4-2A, 4.4-2B and 4.4-2C.

The plant was not specifically designed to meet the requirements of Section XI of the code; therefore, 100 percent compliance may not be feasible or practical. However, access for inservice inspection was considered during the design, and modifications have been made where practical to make provision for maximum access within the limits of the current plant design.

The Reactor Coolant System shall initially be free of gross defects, and the system has been designed such that gross faults or defects should not occur throughout the plant lifetime. The ten-year surveillance program will reveal possible fault areas before any leak develops, should such problems actually occur.

The basis for the surveillance testing at the Reactor Coolant System Pressure Isolation Valves identified in Table TS 3.1-2 is contained within "Order of Modification of License" dated April 20, 1981.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on the general guidance of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam