January 5, 1995

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES (TAC NOS. M88275 AND M88276)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 188 to Facility Operating License No. DPR-58 and Amendment No. 174 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 15, 1993, and supplemented October 7, 1994.

The amendments replace the current TS testing requirements for the Event V reactor coolant system pressure isolation valves with the requirements from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 188 to DPR-58
2. Amendment No. 174 to DPR-74
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\DCCOOK\CO88275.AMD
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#92-276

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NAME	CJamerson		JHickman		CGrimes		EHoller		JHannon
DATE	12/6/94		12/6/94		12/16/94		12/29/94		1/4/95

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DATED: January 5, 1995

AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK-UNIT 1
AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-74-D. C. COOK-UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
J. Hannon
C. Jamerson
J. Hickman (2)
OGC
G. Hill (4)
C. Grimes, O-11F23
P. Campbell
ACRS (4)
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OC/LFDCB
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SEDB

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 5, 1995

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES (TAC NOS. M88275 AND M88276)

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The amendments replace the current TS testing requirements for the Event V reactor coolant system pressure isolation valves with the requirements from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI.

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Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 188 to DPR-58
2. Amendment No. 174 to DPR-74
3. Safety Evaluation

cc w/encl: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 15, 1993 and supplemented October 7, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kevin A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 188

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-17a	3/4 4-17a
3/4 4-17b	--
3/4 4-18*	3/4 4-18*
B 3/4 4-4	B 3/4 4-4

*Overleaf page provided to maintain document completeness. No changes contained on these pages.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator for Fuel Cycle 14,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. Seal line resistance greater than or equal to 2.27E-1 ft/gpm^2 and,

f. The leakage from each Reactor Coolant System Pressure Isolation Valves specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi of the nominal full pressure value.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, declare the leaking valve inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of a combination of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Specification 3.4.6.2.e is applicable with average pressure within 20 psi of the nominal full pressure value.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Determining the seal line resistance at least once per 31 days when the average pressurizer pressure is within 20 psi of its nominal full pressure value. The seal line resistance measured during the surveillance must be greater than or equal to 2.27 E-1 ft/gpm^2. The seal line resistance, R_{SL}, is determined from the following expression:

$$R_{SL} = \frac{2.31\ (P_{CHP} - P_{SI})}{Q^2}$$

where: P_{CHP} = charging pump header pressure, psig

P_{SI} = 2112 psig (low pressure operation)

2262 psig (high pressure operation)

2.31 = conversion factor $(12\ \text{in/ft})^2/(62.3\ \text{lb/ft}^3)$

Q = the total seal injection flow, gpm

The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

TABLE 3.4-0

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE

Valve Number	Valve Size (in.)	Function [a]	Maximum Allowable Leakage (gpm)
SI-170L2	10	ECCS to Reactor Coolant Loop #2 Cold Leg	5
RH 133	8	RHR to Reactor Coolant Loop #2 Cold Leg	4
SI-170L3	10	ECCS to Reactor Coolant Loop #3 Cold Leg	5
RH 134	8	RHR to Reactor Coolant Loop #3 Cold Leg	4

[a] Minimum test differential pressure shall not be below 150 psid.

CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psig, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle 14 will minimize the potential for a large leakage event during steam line break under LOCA conditions. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 12.6 gpm which will limit the calculated offsite doses to within 10 percent of 10 CFR 100 guidelines. Leakage in the intact loops is limited to 150 gpd. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 12.6 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 12.6 gpm.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated November 15, 1993 and supplemented October 7, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 174, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kevin A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 5, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-16a	3/4 4-16a
3/4 4-16b	--
B 3/4 4-4	B 3/4 4-4

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. Seal line resistance greater than or equal to 2.27E-1 ft/gpm^2 and,

f. The leakage from each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi of the nominal full pressure value.

APPLICABILITY: MODES 1, 2, 3 and 4.*

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any reactor coolant system pressure isolation valve(s) leakage greater than the above limit, declare the leaking valve inoperable and isolate the high pressure portion of the affected system from the low pressure portion by the use of at least two closed valves, one of which may be the OPERABLE check valve and the other a closed de-energized motor operated valve. Verify the isolated condition of the closed de-energized motor operated valve at least once per 24 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Specification 3.4.6.2.e is applicable with average pressurizer pressure within 20 psi of the nominal full pressure value.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Determining the seal line resistance at least once per 31 days when the average pressurizer pressure is within 20 psi of its nominal full pressure value. The seal line resistance measured during the surveillance must be greater than or equal to 2.27 E-1 ft/gpm^2. The seal line resistance, R_{SL}, is determined from the following expression:

$$R_{SL} = \frac{2.31\ (P_{CHP} - P_{SI})}{Q^2}$$

where: P_{CHP} = charging pump header pressure, psig

P_{SI} = 2262 psig (high pressure operation)

2.31 = conversion factor $(12 \text{ in/ft})^2/(62.3 \text{ lb/ft}^3)$

Q = the total seal injection flow, gpm

The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation, and

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2. Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5.

TABLE 3.4-0

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVE

Valve Number	Valve Size (in.)	Function [(a)]	Maximum Allowable Leakage (gpm)
SI-170L2	10	ECCS to Reactor Coolant Loop #2 Cold Leg	5
RH 133	8	RHR to Reactor Coolant Loop #2 Cold Leg	4
SI-170L3	10	ECCS to Reactor Coolant Loop #3 Cold Leg	5
RH 134	8	RHR to Reactor Coolant Loop #3 Cold Leg	4

[(a)] Minimum test differential pressure shall not be below 150 psid.

The limitation on seal line resistance ensures that the seal line resistance is greater than or equal to the resistance assumed in the minimum safeguards LOCA analysis. This analysis assumes that all of the flow that is diverted from the boron injection line to the seal injection line is unavailable for core cooling.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Should PRESSURE BOUNDARY LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated November 15, 1993, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. The letter dated November 15, 1993, proposed to delete the list of Event V reactor coolant system pressure isolation valves (PIVs) and the associated surveillance requirements from the technical specifications and test the valves in accordance with the plant's inservice testing program (IST) prescribed by Section 50.55a, "Codes and Standards," of Part 10 of the Code of Federal Regulations (10 CFR 50.55a). In conversations with the staff on September 26, 1994, the licensee agreed to revise the requested change such that rather than deleting the list of valves and the associated surveillance requirements from the technical specifications, a change to the acceptable leakage rates would be proposed. Subsequently, the licensee submitted a revised request by letter dated October 7, 1994. The October 7, 1994, revision clarified the original amendment request by explicitly stating the proposed new requirements as opposed to incorporating them as part of the IST Program. Since the same new requirements would have been implemented by both the original and supplemental submittal, the staff's initial proposed no significant hazards consideration determination did not change. Therefore, renoticing was not warranted.

2.0 BACKGROUND

The Reactor Safety Study (RSS), WASH-1400, identified in a pressurized-water reactor an intersystem loss-of-coolant accident (LOCA) which is a significant contributor to risk of core melt accidents, referred to as "Event V" in the study. The design examined in the RSS contained in-series check valves isolating the high pressure primary coolant from the low pressure injection system piping. The scenario which leads to the Event V accident is initiated

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by the failure of these check valves to function as a pressure isolation barrier. This causes an overpressurization and rupture of the low pressure piping which results in a LOCA that bypasses containment.

To better define the Event V concern, all light-water reactor licensees were requested by NRC's letter dated February 23, 1980, to review the configurations for their high/low pressure interfaces and determine if any Event V configurations existed at their plant(s). By letter dated March 24, 1980, Indiana Michigan Power Company responded to the NRC request. The NRC issued a safety evaluation for an Event V configuration at the D. C. Cook Nuclear Plant, Units 1 and 2, and issued an Order for Modification of License to add the list of PIVs and the associated surveillance requirements to the plant's technical specifications. The specific valve configurations are as follows:

System	Units 1 and 2 Valve Numbers
Low-Head Safety Injection	
Loop 2, cold leg	SI-170L2 and RH133
Loop 3, cold leg	SI-170L3 and RH-134

Each pair of check valves is in series with a normally-closed, motor-operated valve.

The allowed leakage through any reactor coolant system (RCS) PIV is currently limited to 1 gallon per minute (gpm) in Technical Specification 3.4.6.2.f. However, per Technical Specification 3.4.6.2 ACTION c, leakage is allowed to be up to 5 gpm if the most recent measured leakage does not exceed the previous measured leakage by an amount that reduces the margin between the most recent measured leakage and the maximum limit of 5 gpm by 50% or more. The Technical Evaluation Report attached to the NRC's safety evaluation stipulated that when leakage tests are made in cases using pressures lower than function maximum pressure differential, the observed leakage shall be adjusted to "function maximum" pressure differential value. The adjustment is to 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.

Subsequently, the NRC commissioned a study by the Idaho National Engineering Laboratory to survey a number of plants and make recommendations to improve valve leak testing requirements for the Event V valves. The results of the study were compiled in EGG-NTAP-6175, "Inservice Leak Testing of Primary Pressure Isolation Valves," February 1983. One recommendation in the report was that the current technical specification allowances of 1 gpm to 5 gpm leakage should be changed to make the leakage allowance proportional to valve size, based on an allowance of 0.5 gpm per inch nominal valve diameter at the functional differential pressure, with a maximum allowed leakage of 5 gpm, unless it can be shown that the low pressure piping has installed relief capacity such that it would not overpressurize with greater allowed leakage. This recommended allowance has been incorporated into NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," September 1992.

3.0 LICENSEE'S PROPOSED CHANGES AND EVALUATION OF CHANGES

The licensee proposes to change Technical Specification 3.4.6.2(f), which currently specifies the leakage limit of 1 gpm for the reactor coolant system pressure isolation valves listed in Technical Specification Table 3.4-0, to state the following:

> The leakage from each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-0 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System average pressure within 20 psi [pounds per square inch] of the nominal full pressure value.

The resulting allowed leakage as identified in revised Table 3.4-0 is as follows:

SI-170L2	10-inch diameter	5 gpm
SI-170L3	10-inch diameter	5 gpm
RH-133	8-inch diameter	4 gpm
RH-134	8-inch diameter	4 gpm

The Technical Specification action statement will be changed to delete provisions to analyze the leakage of any one of the valves if it is measured between 1 gpm and 5 gpm, and comparing it to the previous leakage, prior to determining the need to declare the valve inoperable. With the proposed change, any valve that exceeds the leakage limit of Technical Specification 3.4.6.2(f) will be declared inoperable and isolated by a closed de-energized motor-operated valve.

In addition, the licensee proposes to delete the specific schedule for performing the leakage surveillance test requirements from the Technical Specification and instead refer to the specification that incorporates the regulatory requirements for performing inservice testing. The current inservice testing program for the D. C. Cook Nuclear Plant, Units 1 and 2, was developed to the requirements of the 1983 Edition, including addenda through Summer 1983 Addenda, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI. The inservice testing program is due to be updated to the requirements of the 1989 Edition of the ASME Code in July 1996. The leakage testing requirements of the later Code edition are essentially the same as the 1983 edition. In accordance with the licensee's inservice testing program, and the requirements of the ASME Code, the leakage testing will be performed once every 2 years and following repair, replacement, or maintenance that could affect the leakage. The Code requirements of paragraph IWV-3420, "Valve Leak Rate Test," include the test frequency, adjustments to the function pressure differential, the methods for measuring the seat leakage, test medium, analysis of leakage rates, and corrective actions.

The corrective action requirements of the ASME Code include provisions for valves 6 inch nominal pipe size and larger such that the previously measured leakage rate is trended and if a higher leakage reduces the margin between the measured leakage rate and the maximum permissible rate by 50% or greater, the test frequency is doubled. Repair or replacement is required if tests show a leakage rate increasing with time, and a projection based on three or more tests indicates that the leakage rate of the next scheduled test will exceed the maximum permissible leakage rate by greater than 10%. Therefore, the proposed Technical Specification change will allow the licensee to monitor a valve with increasing leakage rates rather than declare the valve inoperable. Past experience has shown that higher leakage rates are not always indicative of degradation, and maintenance has been performed based on higher leakage rates unnecessarily. The Code requirements provide an adequate level of quality and safety for monitoring degradation of the valve seats. The change will eliminate unnecessary maintenance and reduce personnel exposure without compromising safety.

The change between the current test schedule and the proposed test schedule is minimal. Because the test can be performed only during plant shutdown conditions, the test will be performed on a refueling outage schedule similar to the current requirement to leak test after each refueling outage. The Code requires leak testing following any repair, replacement, or maintenance that could affect the leakage rate, similar to the current Technical Specification requirement. Finally, the deletion of the TS requirement to perform a leak test during startup whenever the plant has been in cold shutdown for 72 hours or more and if leakage testing has not been performed in the previous 9 months, will actually eliminate very few tests over the life of the plant unless the plant experiences frequent cold shutdowns. The Code requirement to double the test frequency if degrading conditions are indicated will have the same effect as testing every 9 to 12 months during cold shutdown conditions; however, if the valve has not exhibited increasing seat leakage, additional testing will not be required. For as low as is reasonably achievable purposes, increasing the number of tests only when degrading conditions are indicated will eliminate unnecessary personnel dose without impacting the ability to adequately monitor the valves.

4.0 INDIVIDUAL PLANT EXAMINATION RISK ASSESSMENT

The licensee's change request included an assessment of the change in risk based on the Individual Plant Examination study. While the assessment results indicate that the change represents essentially no impact on the risk of operating the plants, the staff did not review the validity or applicability of the licensee's probabilistic risk assessment and did not base its conclusion on the assessment.

5.0 SUMMARY

The proposed changes are essentially equivalent to the current technical specification requirements in that (1) the valves are maintained in the specifications and therefore remain subject to the surveillance requirements of inservice testing and (2) the proposed allowed leakage rates are 5 gpm and 4 gpm, depending on valve size, which are the same or more conservative than the allowed maximum leakage of 5 gpm in the current specifications. The changes also achieve the intent of monitoring the valves for degrading conditions and precluding catastrophic failure by (1) allowing a set limit based on the size of the valve rather than a limit of 1 gpm to 5 gpm, thereby decreasing the potential for valve repairs when there is no substantial degradation, (2) ensuring that radiological exposure to personnel will not be increased, and could be decreased by eliminating maintenance, but requiring additional testing, if only minor increases in leakage are measured, and (3) following the requirements of inservice testing of valves in accord with the regulations and technical specifications. Based on these factors, the proposed technical specification changes are acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (59 FR 623). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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