July 29, 1992

Docket No. 50-315

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. 83190)

The Commission has issued the enclosed Amendment No. 166 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consist of changes to the Technical Specifications (TS) in response to your application dated March 27, 1992 as supplemented by letters dated April 21, May 21, and July 29, 1992.

The amendment changes TS Sections 4.4.5.2, 3.4.6.2, and the Bases 3/4.4.5, 3/4.4.6.2 and 3/4.4.8 to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations. The amendment also reduces the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day to 150 gallons per day. The total allowed primary-to-secondary operational leakage through all steam generators is reduced from one gallon per minute (1440 gallons per day) to .42 gallons per minute (600 gallons per day). This amendment is only applicable for fuel Cycle 13.

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,

/s/
John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

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Enclosures:
1. Amendment No. 166 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

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DATE	07/02/92	07/02/92	7/9/92	7/29/92	7/29/92

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Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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DATED: July 29, 1992

AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK

Docket File
NRC & Local PDRs
PDIII-1 Reading
D.C. Cook Plant File
B. Boger
J. Zwolinski
L. Marsh
M. Shuttleworth
J. Stang
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (8), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
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J. Wiggins 7/D/4
ACRS (10)
GPA/PA
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S. Hoffman 14/B/20
cc: Plant Service list

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
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 March 27, 1992 as supplemented by letters dated April 21, May 21, and July 29, 1992, 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.

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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. 166, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 29, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 166

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 marginal lines indicating the area of change.

REMOVE	INSERT
vi	vi
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
B 3/4 4-2a	B 3/4 4-2a
--------	B 3/4 4-2b
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

INDEX

LIMITING CONDITIONS FOR OPERATIONS & SURVEILLANCE REQUIREMENTS

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

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

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirement of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All tubes that previously had detectable wall penetrations (greater than or equal to 20%) that have not been plugged or repaired by sleeving in the affected area.

*This Specification does not apply in Mode 4 while performing crevice flushing as long as Limiting Conditions for Operation for Specification 3.4.1.3 are maintained.

SURVEILLANCE REQUIREMENTS (Continued)

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for the samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

d. Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle 13) requires a 100% bobbin coil inspection for hot leg tube support plate intersections and cold leg intersections down to the lowest cold leg tube support plate with known outer diameter stress corrosion cracking (ODSCC) indications.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

SURVEILLANCE REQUIREMENTS (Continued)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than or equal to 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

d. Tubes left in service as a result of application of the tube support plate interim plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

3. Degraded Tube or Sleeve means an imperfection greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. Percent Degradation means the amount of the wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit.

6. Repair/Plugging Limit means the imperfection depth at or beyond which the tube or sleeved tube shall be repaired or removed from service. Any tube which, upon inspection, exhibits tube wall degradation of 40 percent or more of the nominal tube wall thickness shall be plugged or repaired prior to returning the steam generator to service. Any sleeve which, upon inspection, exhibits wall degradation of 29 percent or more of the nominal wall thickness shall be plugged prior to returning the steam generator to service. In addition, any sleeve exhibiting any measurable wall loss in sleeve expansion transition or weld zones shall be plugged. This definition does not apply for tubes experiencing outer diameter stress corrosion cracking confirmed by bobbin probe inspection to be within the thickness of the tube support plates. See 4.4.5.4.a.10 for the plugging limit for use within the thickness of the tube support plate.

7. Unserviceable describes the condition of a tube or sleeve if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Inspection determines the condition of the steam generator tube or sleeve from the point of entry (hot leg side) completely

around the U-bend to the top support of the cold leg. For a tube in which the tube support plate elevation interim plugging limit has been applied, the inspection will include all the hot leg intersections and all cold leg intersections down to, at least, the level of the last crack indication.

9. Sleeving a tube is permitted only in areas where the sleeve spans the tubesheet area and whose lower joint is at the primary fluid tubesheet face.

10. The Tube Support Plate Interim Plugging Criteria is used for disposition of a steam generator tube for continued service that is experiencing outer diameter initiated stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging limit, the tube's disposition for continued service will be based upon standard bobbin probe signal amplitude. The plant-specific guidelines used for all inspections shall be amended as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the above voltage/depth parameters. Pending incorporation of the voltage verification requirement in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in the Donald C. Cook Nuclear Plant Unit 1 steam generator inspections for consistent voltage normalization.

    1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 1.0 volt, regardless of the depth of tube wall penetration, if, as a result, the projected end-of-cycle distribution of crack indications is verified to result in primary-to-secondary leakage less than 1 gpm in the faulted loop during a postulated steam line break event. The methodology for calculating expected leak rates from the projected crack distribution must be consistent with WCAP-13187, Rev. 0.

    2. A tube should be plugged or repaired if the signal amplitude of the crack indication is greater than 1.0 volt except as noted in 4.4.5.4.a.10.3 below.

    3. A tube can remain in service with a bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 4.0 volts if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than 4.0 volts will be plugged or repaired.

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging or sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2.

c. Steam generator tube repairs may be made in accordance with the methods described in either WCAP-12623 or CEN-313-P.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The results of inspections performed under 4.4.5.2 for all tubes in which the tube support plate interim plugging criteria has been applied shall be reported to the Commission within 15 days following the inspection. The report shall include:

1. Listing of applicable tubes.

2. Location (applicable intersections per tube) and extent of degradation (voltage).

e. The results of steam line break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the Commission prior to restart for Cycle 13.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[2]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The third and fourth steam generators not inspected during the first inservice inspection shall be inspected during the second and third inspections, respectively. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTI

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or sleeve defective tubes and inspect additional 4S tubes in this S.G.	C-1 C-2	None Plug or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or sleeve defective tubes, and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
		Prompt notification to NRC pursuant to specification 6.9.1	Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or sleeve defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

S = 3 (N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One of the containment atmosphere particulate radioactivity monitoring channels (ERS-1301 or ERS-1401),

b. The containment sump level and flow monitoring system, and

c. Either the containment humidity monitor or one of the containment atmosphere gaseous radioactivity monitoring channels (ERS-1305 or ERS-1405).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring channels are inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months,

c. Containment humidity monitor (if being used) - performance of CHANNEL CALIBRATION at least once per 18 months.

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 13,

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. 1 GPM leakage from any reactor coolant system pressure isolation valve specified in Table 3.4-0.

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, except when:

1. The leakage is less than or equal to 5.0 gpm, and

2. The most recent measured leakage does not exceed the previous measured leakage* by an amount that reduces the

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* To satisfy ALARA requirements, measured 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.

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

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY

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 a modification 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 generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system. The allowable primary-to-secondary leak rate is 150 gallons per day per steam generator for one fuel cycle (Cycle 13). Axial or circumferentially oriented cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Leakage in excess of this limit will require plant shutdown and an inspection, during which the leaking tubes will be located and plugged or repaired. A steam generator while undergoing crevice flushing in Mode 4 is available for decay heat removal and is operable/operating upon reinstatement of auxiliary or main feed flow control and steam control.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the repair limit which is defined in Specification 4.4.5.4.a. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates are plugged or repaired by the criteria of 4.4.5.4.a.10.

3/4.4.5 STEAM GENERATORS TUBE INTEGRITY (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

BASES

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Fuel Cycle 13 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 1 gpm. This is less than the 120 gpm used to calculate the offsite doses 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 1 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 1 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.

The Surveillance Requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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.

BASES

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Cook Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

Offsite doses following a main steam line break are limited to 10 percent of the 10 CFR 100 guideline. The restriction is based on a Cook Nuclear Plant site-specific radiological evaluation that assumes a post-accident primary-to-secondary leak rate of 120 gpm in the faulted loop and a primary coolant specific activity concentration corresponding to 1% fuel defects (approximately 4.6 microCuries/gram dose equivalent I-131), rather than a specific activity of 1.0 microCuries dose equivalent I-131.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NOS. 50-315

1.0 INTRODUCTION

By letter dated March 27, 1992 as supplemented by letters dated April 21, May 21, and July 29, 1992, the Indiana Michigan Power Company (the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendments would change TS Sections 4.4.5.2, 3.4.6.2, and the Bases 3/4.4.5, 3/4.4.6.2 and 3/4.4.8 to allow the implementation of interim steam generator tube plugging criteria for the tube support plate elevations. The amendment also reduces the allowed primary-to-secondary operational leakage from any one steam generator from 500 gallons per day to 150 gallons per day. The total allowed primary-to-secondary operation leakage through all steam generators is reduced from one gallon per minute (1440 gallons per day) to .42 gallons per minute (600 gallons per day). This amendment is only applicable for fuel Cycle 13.

2.0 BACKGROUND

Previous inservice inspections and examinations of the steam generator (SG) tubes at D. C. Cook Unit 1 have identified intergranular stress corrosion cracking (IGSCC) on the outer diameter of the tubes at the tube support plate (TSP) intersections. The licensee refers to this particular form of IGSCC as outer diameter stress corrosion cracking (ODSCC).

Outer diameter stress corrosion cracking activity at TSP intersections is a common degradation phenomenon in SGs in nuclear power plants. Approximately 21 tubes, including 57 tube-to-TSP intersections, have been removed from affected SGs across the industry for examination and testing. These include 3 tubes from Cook Unit 1 (including 3 TSP intersections) and 12 tubes from the removed Cook Unit 2 (21 TSP intersections). Each of these pulled tube TSP intersections was sectioned and metallographically examined. In general, these examinations have revealed multiple, segmented, and axial cracks with short lengths for the deepest penetrations. The outer diameter stress corrosion cracking is generally confined to within the thickness of the TSPs, consistent with the corrosion mechanism which involves the concentration of impurities, including caustics, in the tube-to-TSP crevices. The staff notes that there is some potential for shallow ODSCC for a short distance above or below the TSP. This has been observed for 2 of the pulled TSP intersections from another plant.

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The pulled tube specimens from Cook Unit 1 to date have shown minimal intergranular attack (IGA) involvement with the ODSCC. However, more significant IGA involvement has been observed on some pulled tube specimens from other plants. These results suggest that the degradation develops as IGA plus stress corrosion cracking (SCC), particularly when maximum IGA depths greater than 25% are found. A large number (>100) of axial cracks around the circumference are commonly found on these tubes. The maximum depth of IGA is typically 1/2 to 1/3 of the SCC depth. Patches of cellular IGA/ODSCC formed by combined axial and circumferential orientation of microcracks are frequently found in pulled tube examinations. The staff notes, however, that the axial crack segments have been the dominant flaw feature affecting the structural integrity of the pulled tube specimens as evidenced by results of burst tests (discussed in Section 4.3) performed for 29 of the pulled TSP intersections prior to sectioning.

Technical Specification 4.4.5.4.a.6, Plugging or Repair Limit, requires that tubes with imperfections exceeding 40% of the nominal tube wall thickness be repaired by sleeving or removed from service by plugging. The licensee stated that this repair criterion would result in unnecessary removal of significant numbers of SG tubes from service. To preclude this, the licensee developed proposed alternative plugging criteria (APC) that was submitted by letter dated March 20, 1992 (Reference 3). The proposed APC involves a voltage amplitude limit of four volts, as measured by the industry standard eddy current bobbin coil probe (referred to herein as a bobbin) using the 400/100 KHz mix differential channel, in lieu of the current 40% depth-based plugging or repair limit. These criteria would only apply to ODSCC degradation confined to within the thickness of the TSPs.

In their April 21, 1992 letter (Reference 2), the licensee requested interim modifications to the tube repair limits and primary-to-secondary leakage limits in the Technical Specifications for the 13th operating cycle only, pending completion of the staff's review of the APC proposal. The proposed modifications to the tube repair limits are described in detail in Section 3.0 of this Safety Evaluation, and include a one volt repair criterion for flaws confined to the thickness of the TSP in lieu of the currently applicable depth-based limit of 40%. The proposed modifications to the leakage limits, described in Section 4.0 of this Safety Evaluation, are more restrictive than the present limits.

3.0 TECHNICAL SPECIFICATION CHANGES

Cook Unit 1 Technical Specification 4.4.5.4.a.6, Plugging or Repair Limit, and Bases 3/4.4.5, Steam Generators, are revised to specify that the repair limit at the TSP intersections for the 13th operating cycle is based on the analysis in WCAP-13187, Revision 0 (Reference 4), to maintain SG tube serviceability as described below:

a. An eddy current inspection using a bobbin of 100% of the hot leg TSP intersections and down to the lowest cold leg SG TSP intersections with known outside diameter stress corrosion cracking will be performed for tubes in service.

b. Degradation within the bounds of the TSP with a bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.

c. Degradation within the bounds of the TSP with a bobbin voltage greater than 1.0 volt will be repaired or plugged except as noted in d. below.

d. Indications of potential degradation within the bounds of the TSP with a bobbin voltage greater than 1.0 volt, but less than or equal to 4.0 volts, may remain in service if a rotating pancake coil probe (RPCP) inspection does not detect degradation. Indications of degradation with a bobbin voltage greater than 4.0 volts will be plugged or repaired.

Cook Unit 1 Technical Specification 3.4.6.2 and Bases 3/4.4.5 are revised to specify that, for the thirteenth operating cycle only, primary-to-secondary leakage through all SGs shall be limited to 600 gpd and 150 gpd through any one SG. Primary-to-secondary leakage during a steamline break (SLB) will not exceed the current Technical Specification basis of one gpm.

4.0 EVALUATION

4.1 Inspection Issues

In support of the proposed interim repair limit, the licensee proposes to utilize the eddy current test guidelines provided in Attachment 4 of the March 20, 1992, letter (Reference 3) to ensure that the field bobbin indication voltage measurements are obtained in a manner consistent with how the voltage limit was developed. These guidelines define the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used in the inspection of the steam generators. To supplement these guidelines, the licensee will also record all voltage indications less than 1.0 volt as referenced in Attachment 2 of the May 21, 1992, letter (Reference 5). All flaw indications, regardless of voltage amplitude, will be recorded. With these commitments, the staff finds the licensee's eddy current test guidelines to be acceptable.

The staff finds that the proposed bobbin inspection program is consistent with the development of voltage-based repair limits; namely, the establishment of the relationship between burst pressure and bobbin voltage. In addition, the licensee states in its May 21, 1992, Attachment 2 (Reference 5) that it will perform an RPCP sample inspection of tubes at TSP intersections. The program will include dents greater than five volts as measured by the bobbin probe and TSP intersection artifact indications or indications with unusual phase angles. This sample program will be performed on up to about 100 tube intersections. The RPCP can provide improved resolution of flaw indications as compared to bobbin probe in the presence of dents and artifacts and is sensitive to both axial and circumferential flaws. The licensee states that the sampling program will be expanded as necessary, based on the nature and number of flaws discovered. In addition, tubes in the RPCP sample program that are found to have RPCP flaw indications will be plugged or repaired.

The Cook Nuclear Plant Unit 1 eddy current test guidelines in Attachment 4 of Reference 3 are supplemented by Attachment 2 of the licensee's letter of May 21, 1992, to require the RPCP inspection of TSP intersections exhibiting bobbin indications exceeding 1.0 volt. The RPCP inspections will permit better characterization of the indications found by the bobbin to confirm or deny the existence of any actual tube degradation. The proposed repair limit is based on the axially oriented ODSCC as the dominate degradation mechanism with some IGA involvement. The proposed limit is also based on the premise that any significant degradation is confined to the TSP. The licensee has agreed to inform the staff prior to Cycle 13 operation of any unforeseen RPCP findings relative to the characteristics of the flaws at the TSPs. This includes any detectable circumferential indication or detectable indications extending outside the thickness of the TSP. A safety evaluation of these unforeseen findings will also be provided.

4.2 Tube Integrity Issues

The purpose of the Technical Specification tube repair limits is to ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria 14, 15, 31 and 32 of 10 CFR Part 50, Appendix A. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubing. Leakage integrity refers to limiting primary-to-secondary leakage to within acceptable limits. The traditional strategy for accomplishing these objectives has been to establish a minimum wall thickness requirement in accordance with the structural criteria of Regulatory Guide 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Allowance for eddy current measurement error and flaw growth between inspections has been added to the minimum wall thickness requirements (consistent with the Regulatory Guide) to arrive at a depth-based repair limit. Enforcement of a minimum wall thickness requirement would implicitly serve to ensure leakage integrity (during normal operation and accidents), as well as structural integrity. It has been recognized, however, that defects, especially cracks, will occasionally grow entirely through-wall and develop small leaks. For this reason, tight limits on allowable primary-to-secondary leakage have been established in the Technical Specifications to ensure timely plant shutdown before adequate structural and leakage integrity of the affected tube is impaired.

The proposed tube repair limits for Cook Unit 1 consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the proposed repair criteria represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The pulled tube examinations show that for bobbin indications at or near 1 volt (i.e., the proposed interim limit) maximum crack depths range between 20% and 98% through-wall. The likelihood of through-wall or near through-wall crack penetrations appears to increase with increasing voltage amplitude. For indications at or near 4.0 volts (i.e., the APC limit), the maximum crack depths have been found to range between 90% and 100% through-wall. Clearly, many of the tubes which will be found to contain "non-repairable" indications

under the proposed interim criteria may develop through-wall and near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated SLB accidents. The staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in the Safety Evaluation in Sections 4.3 and 4.4, respectively.

4.3 Structural Integrity

4.3.1 Burst Integrity

The licensee has developed a burst strength/voltage correlation to demonstrate that bobbin indications satisfying the proposed 1.0 volt interim repair criterion or the 4.0 volt APC criterion will retain adequate structural margins during Cycle 13 operation, consistent with the criteria of Regulatory Guide 1.121. The burst strength/voltage correlation includes the burst pressure versus field bobbin voltage data (pre-pull values) for 29 pulled tubes including 3 TSPs from Cook Unit 1. This pulled tube data are supplemented by 30 data points from laboratory tube specimens containing ODSCC flaws produced in model boiler tests under simulated field conditions. The bobbin voltage data used to construct the burst pressure/voltage correlation have been normalized to reflect calibration standard voltage set-ups and voltage measurement procedures consistent with the NDE Data Acquisition and Analysis Guidelines in WCAP-13187, Revision 0, Appendix A (Reference 4). The staff finds that this normalization ensures consistency among the voltage data in the burst pressure/voltage correlation and, in addition, ensures consistency between the voltage data in the correlation and the field voltage measurements at Cook Unit 1.

The most limiting burst pressure criterion of Regulatory Guide 1.121 is that degraded tubes shall retain a margin of three against burst at normal operating differential pressure across the tube. For Cook Unit 1, this translates to a limiting burst pressure criterion of 4275 psi. From the burst pressure/voltage correlation, the maximum voltage which will satisfy this burst pressure criterion at a 95% confidence intervals is 6.8 volts. The 4.0 volt APC limit, which WCAP-13187 is intended to support, includes an allowance for 20% NDE measurement uncertainty and for a 50% increase in voltage during the next operating cycle. The NDE measurement uncertainty estimate considered measurement uncertainties stemming from bobbin design characteristics, bobbin wear (which affects centering), variability among American Society of Mechanical Engineers (ASME) calibration standards, and variability in the analysts' interpretation of the signal voltage. The staff concurs that the NDE Data Acquisition and Analysis Guidelines (Reference 4, Appendix A), which have been incorporated into the Cook Unit 1, eddy current test guidelines, will be effective in minimizing the uncertainties as they apply to the interim criteria. Based on implementation of these guidelines, a cumulative probability distribution of the residual measurement uncertainty (applicable to each bobbin indication) has been developed. The assumed 20% uncertainty in the voltage measurements is conservative with respect to the upper 90% cumulative probability value of 16% as determined from the cumulative probability distribution.

Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during Cycles 10 and 11 at Cook Unit 1.

Specifically, the eddy current data from the 1987 and 1989 inspections were re-examined for each indication reported during the most recent previous inspection in 1990 using a consistent data analysis procedure. This examination showed that nearly all of the 1990 indications were traceable back to the 1987 and 1989 inspections. The average percent changes in voltage, considering the entire data set, were 44% between 1987 and 1989 and 45% between 1989 and 1990. These averages conservatively treat negative voltage changes as zero changes. If the data set is restricted to voltage changes where the initial indication exceeded 0.75 volts, the average voltage changes are smaller, e.g., 8% between 1989 and 1990. The 50% average voltage growth allowance used to support the 4.0 volt APC limit is intended to provide margins for variation in future growth rates at Cook Unit 1.

For any specific individual tube, NDE measurement uncertainty and/or voltage growth may exceed the values assumed in the above deterministic basis for the 4.0 volt APC repair limit, since the deterministic basis does not consider the tail of the voltage measurement and voltage growth distribution. In addition, burst pressure for some tubes may be less than the 95% confidence values in the burst pressure/voltage calculation. The licensee proposes that these uncertainties be directly accounted for by use of Monte Carlo methods to demonstrate that the probability of burst during SLB accidents is acceptably low for the distribution of voltage indications being left in service. Under this approach, the beginning-of-cycle (BOC) indications left in service are projected to the end-of-cycle (EOC) by randomly sampling the probability distributions for NDE uncertainties and voltage growth per cycle. For each EOC Monte Carlo sample of bobbin voltage, the burst pressure/voltage correlation is randomly sampled to obtain a burst pressure. The 100,000 Monte Carlo samples are performed for the BOC distribution. The probability of tube burst at SLB is obtained as the sum of the samples resulting in burst pressures less than the SLB pressure differential of 2650 psi divided by the number of times the distribution of indications left in service is sampled.

This kind of Monte Carlo analysis was performed for the distribution of indications found during the previous (i.e., 1990) inspection at Cook Unit 1. This analysis indicated that implementation of a 4.0 volt repair criterion at that time would have yielded a conditional probability of burst, given an SLB of 1×10^{-5}. The staff concurs that this is an extremely low probability, three orders of magnitude less than the value considered in a staff generic risk assessment for SGs (NUREG-0844). Over time, the number of indications found between 0 and 4.0 volts can be expected to increase.

Therefore, the APC proposal (involving the 4.0 volt repair criterion) includes a provision for determining the probability of burst at SLB conditions following each outage for indications left in service to confirm the continued adequacy of the repair criterion.

The staff is continuing to evaluate the technical basis for the proposed APC (i.e., 4.0 volt criterion). In the meantime, the staff concludes that the proposed 1.0 volt interim criterion will provide adequate assurance that tubes

with indications which are accepted for continued service will meet the burst pressure criteria of Regulatory Guide 1.121. The staff notes that the bounding value of voltage growth/cycle at Cook Unit 1, since 1987, has not exceeded 0.8 volts. The staff estimates this 0.8 volts to represent a bounding value, assuming no increase in corrosion rates over what has been observed previously at Cook Unit 1. Assuming a 20% voltage measurement uncertainty (upper 98% confidence value determined by the licensee) for a 1.0 volt indication left in service, the EOC voltage is expected by the staff to be bounded by 4.0 volts. This is substantially below the 6.8 voltage limit evaluated by the licensee as the lower 95% confidence limit for meeting the most limiting burst pressure criterion (i.e., three times normal operating pressure differential).

Finally, the licensee is proposing as part of the interim repair criteria that indications with bobbin voltages greater than 1.0 volt, but less than or equal to 4.0 volts, remain in service if RPCP inspection does not confirm the indication. The staff notes that short and/or relatively shallow cracks that are detectable by the bobbin may sometimes not be detectable by RPCP, although the RPCP is considered by the staff to be more sensitive to longer, deeper flaws which are of structural significance.

The staff further notes that burst strength is not a unique function of voltage, rather for a given voltage there is a statistical distribution of possible burst strengths as indicated in the burst pressure/voltage correlation. The staff concludes that burst pressures for bobbin indications which were not confirmed by RPCP will tend to be at the upper end of the burst pressure distribution. The 4.0 volt cutoff, such that all bobbin indications would be plugged or repaired (with or without confirming RPCP indications), provides additional assurance that all excessively degraded tubes will be removed from service. Thus, the staff finds the proposed exception to the 1.0 volt criterion to be acceptable.

4.3.2 Combined Accident Loadings

The licensee has evaluated the effects of combined safe shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) loads and SSE plus SLB loads on tube integrity, consistent with the General Design Criterion 2 (GDC-2) of 10 CFR Part 50, Appendix A. A combined LOCA plus SSE must be evaluated for potential yielding of the TSPs which could result in subsequent deformation of the tubes. If significant tube deformation should occur, primary flow area could be reduced and postulated cracks in tubes could open up which might create the potential for in-leakage (i.e., secondary-to-primary) under LOCA conditions. In-leakage during LOCA would pose a potential concern since it may cause an increase in the core peak clad temperature (PCT).

The most limiting accident conditions for tube deformation considerations result from the combination of SSE and LOCA loads. The seismic excitation defined for SGs is in the form of acceleration response spectra at the SG supports. In the seismic analysis, the licensee has used generic response spectra which envelop the Cook-specific response spectra. A finite element model of the Series 51SG was developed and the analysis was performed using

the WECAN computer program. The mathematical model consisted of three dimensional lumped mass, beam, and pipe elements as well as general matrix input to represent the piping and support stiffness. Interactions at the TSP shell and wrapper/shell connections were represented by concentric spring-gap dynamic elements. Impact damping was used to account for energy dissipation at these locations.

Prior qualification of the Cook Unit 1 primary piping for leak-before-break requirements resulted in the limiting LOCA event being the break of a minor branch line. The licensee, however, has used the loads for the primary piping break as a conservative approximation. The principal tube loading during a LOCA is caused by the rarefaction wave in the primary fluid. This wave initiates at the postulated break location and travels around the tube U-bends. A differential pressure is created across the two legs of the tube which causes an in-plant horizontal motion of the U-bends and induces significant lateral loads on the tubes. The pressure time histories needed for creating the differential pressure across the tube are obtained from transient thermal-hydraulic analyses using the MULTIFLEX computer code. For the rarefaction wave induced loadings, the predominant motion of the U-bends is in the plane of the U-bend. Thus, the individual tube motions are not coupled by the anti-vibration bars and the structural analysis is performed using single tube models limited to the U-bend and the straight leg region over the top two TSPs.

In addition to the rarefaction wave loading discussed above, the tube bundle is subjected to bending loads during a LOCA. These loads are due to the shaking of the SG caused by the break hydraulics and reactor coolant loop motion. However, the resulting TSP loads from this motion are small compared to those due to the rarefaction wave induced motion.

To obtain the LOCA induced hydraulic forcing functions, a dynamic blowdown analysis is performed to obtain the system hydraulic forcing functions assuming an instantaneous (1.0 msec break opening time), double-ended guillotine break. The hydraulic forcing functions are then applied, along with the displacement time-history of the reactor pressure vessel (obtained from a separate reactor vessel blowdown analysis) to a system structural model that includes the SG, the reactor coolant pump, and the primary piping. This analysis yields the time-history displacements of the SG at its upper lateral and lower support nodes. These time-history displacements formulate the forcing functions for obtaining the tube stresses due to LOCA shaking of the SG.

In calculating a combined TSP load, the licensee combined the LOCA rarefaction and LOCA shaking loads directly, while the LOCA and SSE loads were combined using the square root of the sum of the squares. The staff found this combination methodology acceptable. The overall TSP load was transferred to the SG shell through wedge groups located at discrete locations around the plate circumference.

The radial loads due to combined LOCA and SSE could potentially result in yielding of the TSP at the wedge supports, causing some tubes in the vicinity of the wedge supports to be deformed. Utilizing results from recent tests and

analysis programs, the licensee has shown that tubes will undergo permanent deformation if the change in diameter exceeds a minimum threshold value. This threshold for tube deformation is related to the concern for tubes with pre-existing tight cracks that could potentially open during a combined LOCA plus SSE event. For Cook Unit 1, the LOCA plus SSE loads were determined to be of such magnitude that none of the tubes (which are assumed to contain pre-existing tight cracks) are predicted to exceed this deformation threshold value and, therefore, will not lead to significant tube leakage.

The licensee has assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks. Tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously performed tests, the licensee has concluded that the burst strength of tubes with through-wall cracking is not affected by an SSE event.

Based on a review of the information provided by the licensee, the staff concludes that at Cook Unit 1, no significant tube deformation or leakage is likely to occur during an SSE plus LOCA event. In addition, burst strength of tubing with through-wall cracks is not affected by an SSE event.

4.4 Leakage Integrity

As discussed earlier, a number of the indications satisfying the proposed interim 1.0 volt repair limit can be expected to have or to develop through-wall and/or near through-wall crack penetrations during the next cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. The staff finds that adequate leakage integrity during normal operating conditions is assured by the proposed restrictive Technical Specification limits on allowable primary-to-secondary leakage as discussed in Section 4.5 of this Safety Evaluation. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that, for the most limiting accident, the resulting leakage will not exceed the rate assumed in the Cook Unit 1 design basis accident.

The licensee has identified in the Final Safety Analysis Report (FSAR), Chapter 14, accidents which result in secondary steam release, and thus whose consequences could be affected by the extent of primary-to-secondary leakage. Of these accidents, the SLB was determined to be the most limiting. In this case, since the SG in the faulted loop is subject to dryout, the activity release path is conservatively assumed to be direct to the environment, without any mitigation resulting from mixing with secondary liquid coolant in the SG.

The licensee has submitted an analysis (Reference 4) in support of the APC 4.0 volt based criteria proposal to demonstrate that 120 gpm as the total allowable primary-to-secondary leakage rate during SLB is not exceeded. This analysis is under staff review. For the purpose of supporting the interim repair limit proposal, the licensee has proposed that the maximum allowable primary-to-secondary leak rate during SLB be 1.0 gpm, which is consistent with the assumed leak rate in the FSAR design basis analysis. Therefore, there is

no change in the off-site dose as a result of the use of the interim repair limit. The staff concurs that use of 1.0 gpm is an acceptable primary-to-secondary leak rate limit and is well bounded by the maximum 120 gpm primary-to-secondary allowable during an SLB event.

The SLB leakage calculation model in Reference 4 utilizes a correlation between leakage test data obtained under simulated SLB conditions (at a given TSP location), and the corresponding normalized bobbin voltage (SLB leakage/voltage correlation). The SLB leakage data includes 27 data points from the model boiler specimens described earlier and 7 data points from the pulled tube specimens. The calculation method involves establishing the voltage distribution of the indications being accepted for continued service. Probability distributions of voltage measurement uncertainty, voltage growth/cycle, and SLB leak rate versus voltage are accounted for by Monte Carlo techniques in predicting the distribution of EOC voltages and the associated SLB leakage. One thousand Monte Carlo simulations of the BOC distribution of indications are performed. SLB leakage is evaluated at the 90% cumulative probability level.

Based on the voltage distributions found during previous inspection (1990) at Cook Unit 1, and assuming implementation of the 4.0 volt repair criterion, the estimated leakage during a postulated SLB at EOC 12 was 0.1 gpm at the 90% cumulative probability level, well within the current licensing basis of 1.0 gpm. Utilizing the same model, the licensee has determined that about 4000 BOC indications at 2 volts would be necessary to produce an SLB leak rate of 1.0 gpm (at 90% cumulative probability) at EOC.

In support of the one volt interim repair criterion, the licensee will update the above analysis to consider the distribution of voltages for indications satisfying the one volt criterion during the thirteenth refueling outage inspection if voltages deviate from the original simulation and, therefore, are not bounded by the original analysis. The analysis will also reflect the distribution of voltage changes observed during Cycle 12 (i.e., 1990 to 1992).

In addition to the above analysis at the staff's request by letter dated July 29, 1992, the licensee committed to also verify the 1.0 gpm at EOC for SLB using a deterministic calculation method. The method will consist of the following:

- Determine the end-of-cycle (EOC) voltage distribution in terms of the number of indications falling into each of the following EOC voltage ranges:

  $\leq$ 2.5 volts
  > 2.5 to 4 volts
  > 4 volts

- Acceptable methods for determining the EOC voltage distribution include:

  - The methodology described in WCAP-13187. (This involves sampling of the cumulative probability distributions of NDE measurement error and

of voltage growth during the most recent operating cycle using Monte Carlo techniques and applying the results to the beginning-of-cycle (BOC) voltage distribution.)

- A simplified approach may be used as an alternative (to the WCAP-13187 approach) provided it provides for a conservative treatment of the tails of the cumulative probability distributions of NDE measurement error and of voltage growth to the 100% cumulative probability values.

- SLB leakage as a function of EOC voltage shall be determined as follows:

EOC Voltage	SLB Leakage
≤ 2.5 volts	0
> 2.5 to 4 volts	1 liter/hour
> 4 volts	10 liters/hour

The staff's approval of the proposed interim repair limit is based on the licensee's being able to demonstrate that acceptance of all bobbin indications satisfying the 1.0 volt criterion will not create the potential for leakage in excess of the 1.0 gpm licensing basis for a postulated SLB. By letters dated May 21, 1992, (Reference 5) and July 29, 1992, (Reference 6) the licensee has agreed to submit reports on the results of the SLB leakage analysis prior to restart for Cycle 13.

4.5 Proposed Interim Leakage Limits

4.5.1 Description

The licensee is proposing an interim change to the reactor coolant system leakage limit criteria in Technical Specification 3.4.6.2 that is applicable to the thirteenth operating cycle only. Specifically, the licensee is proposing to reduce the current 500 gpd limit for primary-to-secondary leakage through any one SG to 150 gpd. In addition, the limit on total leakage through all SGs would be reduced from 1.0 gpm (1440 gpd) to .42 gpm (600 gpd). Leakage during an SLB would not exceed the current design basis of 1.0 gpm.

4.5.2 Discussion

The current 500 gpd limit per SG is intended to ensure that through-wall cracks which leak at rates up to this limit during normal operation will not propagate and result in tube rupture under postulated accident conditions consistent with the criteria of Regulatory Guide 1.121. The current 1.0 gpm limit for total primary-to-secondary leakage is consistent with the assumptions used in the FSAR design basis accident analyses.

Development of the proposed 150 gpd limit per SG has utilized the extensive industry data base regarding burst pressure as a function of crack length and leakage during normal operation. Based on leakage evaluated at the lower 95% confidence interval for a given crack size, the proposed 150 gpd limit would

be exceeded before the crack length reaches the critical crack length for SLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical length for three times normal operating pressure.

The proposed interim change is more restrictive than the existing limits and is intended to provide a greater margin of safety against rupture. The proposed interim limits are also intended to provide an additional margin to accommodate a rogue crack which might grow at much greater than expected rates, or unexpectedly extend outside the thickness of the TSP, and thus provide additional protection against exceeding SLB leakage limits. The staff finds the proposed interim leakage limits to be acceptable.

4.6 Summary

Based on the above evaluation, the staff concludes that the proposed interim tube repair limits and leakage limits will ensure adequate structural and leakage integrity of the SG tubing at Cook Unit 1, consistent with applicable regulatory requirements. The staff's approval of the proposed interim repair limit is based on the licensee's being able to demonstrate that acceptance of all indications satisfying the 1.0 volt criterion will not create the potential for leakage in excess of the 1.0 gpm licensing basis for a postulated SLB occurring at EOC 13.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 24517). Accordingly, the amendment meets 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 amendment.

7.0 CONCLUSION

The Commission 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Stang
E. Murphy

Date: July 29, 1992

REFERENCES

1. Indiana Michigan Power Company letter, AEP:NRC:1166A, dated March 27, 1992, "Donald C. Cook Nuclear Plant - Unit 1, Steam Generator (SG) Tube Support Plate (TSP) Interim Plugging Criteria (IPC)."

2. Indiana Michigan Power Company letter, AEP:NRC:1166B, dated April 21, 1992, "Donald C. Cook Nuclear Plant - Unit 1, Steam Generator Tube Support Plate Interim Plugging Criteria."

3. Indiana Michigan Power Company letter, AEP:NRC:1166, dated March 20, 1992, "Donald C. Cook Nuclear Plant - Unit 1, Steam Generator (SG) Tube Support Plate (TSP) Alternate Plugging Criteria (APC)."

4. Westinghouse Reports WCAP-13187, Revision 0, (Proprietary Version) and WCAP-13188, Revision 0 (Non-Proprietary Version), "Donald C. Cook Nuclear Plant - Unit 1, SG Tube Plugging Criteria for Indications at Tube Support Plates."

5. Indiana Michigan Power Company letter, AEP:NRC:1166C, dated May 21, 1992, "Donald C. Cook Nuclear Plant Unit 1, Modification to Technical Specification Change Request to Allow Interim Plugging Criteria of 1.0 Volt."

6. Indiana Michigan Power Company letter, AEP:NRC:1166D, dated July 29, 1992, "Donald C. Cook Nuclear Plant Unit 1, Modification to Technical Specification change Request to Allow Interim Plugging Criterion of 1.0 Volt."