March 15, 1994

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 43215

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: INCORPORATION OF 2.0 VOLT STEAM GENERATOR TUBE SUPPORT PLATE INTERIM PLUGGING CRITERIA FOR CYCLE 14 (TAC NO. M85971)

The Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 15, 1993, as supplemented February 15 and 24, 1994.

The amendment modifies the TS to incorporate 2.0 volt steam generator tube support plate interim plugging criteria for cycle 14.

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,

Original signed by:

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

Enclosures:
1. Amendment No. 178 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

A few apparent typos. Please fix them (w APH

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Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

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December 1993

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. 178
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 December 15, 1993, as supplemented February 15, 1994, and February 24, 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.

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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. 178 , 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

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: March 15, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 178

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 4-8	3/4 4-8
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-16	3/4 4-16
B 3/4 4-2a	B 3/4 4-2a
B 3/4 4-4	B 3/4 4-4

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. In addition to the sample required in 4.4.5.2.b.1 through 3, all tubes which have had the F* criteria applied will be inspected in the roll expanded region. The roll expanded region of these tubes may be excluded from the requirements of 4.4.5.2.b.1.

d. 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.

e. Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle 14) 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.

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 requirements 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 2.0 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 12.6 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, and as prescribed in draft NUREG-1477.

    2. A tube should be plugged or repaired if the signal amplitude of the crack indication is greater than 2.0 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 2.0 volt but less than or equal to 3.6 volts if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than 3.6 volts will be plugged or repaired.

11. F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.11 inches (not including eddy current uncertainty).

12. F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

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 or that have defects below the F* distance and were not plugged 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 14.

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. 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 allowabe primary-to-secondary leak rate is 150 gallons per day per steam generator for one fuel cycle (Cycle 14). 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.

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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letters dated December 15, 1993, February 15, 1994, and February 24, 1994, Indiana Michigan Power Company, the licensee, submitted a request to change the Technical Specifications (TS) for Donald C. Cook Nuclear Plant, Unit No. 1. The requested amendment revises, in part, TS 4.4.5 and 3.4.6.2 and Bases 3/4.4.5 and 3/4.4.6.2 to allow the continuance of voltage-based steam generator tube plugging criteria for defects located at the tube support plate elevations. All of the proposed changes are applicable to the fourteenth operating cycle only.

The proposed voltage criterion pertains specifically to outside diameter stress corrosion cracking (ODSCC) flaws, and the proposed criterion (1) permits flaws within the bounds of the tube support plate elevations with bobbin voltages less than or equal to 2.0 volts to remain in service, (2) permits flaws within the bounds of the tube support plate with bobbin voltages greater than 2.0 volts but less than 3.6 volts to remain in service if a rotating pancake coil (RPC) probe does not detect degradation, and (3) requires flaw indications at the tube support plate elevations with bobbin voltages greater than 3.6 volts to be plugged or repaired.

The staff is currently developing a generic interim position on voltage-based limits for ODSCC at tube support plate elevations. The staff has published several tentative conclusions regarding voltage-based plugging criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes"; however, the staff is continuing to evaluate an acceptable generic position which takes into consideration public comments received on draft NUREG-1477, domestic operating experience under the voltage-based repair criteria, and additional data which has been made available from European nuclear power plants. The staff currently plans to document its final position in a generic letter with the disposition of public comments being documented in the final version of NUREG-1477. In the meantime, pending completion and issuance of the staff's generic position on the voltage-based interim plugging criteria (IPC), the staff is continuing to evaluate IPC proposals on a case-specific basis, as necessary, to ensure that there is adequate assurance of public health and safety.

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2.0 BACKGROUND

By letter dated March 20, 1992, Indiana Michigan Power Company submitted a steam generator tube support plate alternate plugging criteria (APC) request. As a result of technical issues raised during the review of this and other similar submittals, the full APC repair limit was not approved by the NRC staff; however, a reduced IPC repair limit was approved on a one-cycle basis. The modifications to the tube repair limits for D.C. Cook Unit No. 1 as a result of this IPC approval were documented in an amendment package dated July 29, 1992, "Donald C. Cook Nuclear Plant, Unit 1 - Amendment No. 166 to Facility Operating License No. DPR-58 (TAC No. M83190)" (Reference 1). The tube repair limits documented in Reference 1 included a 1.0 volt repair criterion for ODSCC flaws confined to within the thickness of the tube support plate in lieu of the depth-based limit of 40-percent. In addition, Reference 1 allowed bobbin indications between 1.0 and 4.0 volts to remain in service provided RPC inspection of these indications did not confirm the degradation to be present. The staff concluded in Reference 1 that the proposed interim tube repair limits and leakage limits would ensure adequate structural and leakage integrity of the steam generator tubing at Donald C. Cook Nuclear Plant Unit No. 1, consistent with applicable regulatory requirements, for the thirteenth operating cycle. The licensee's current proposal is applicable to cycle fourteen operation and is similar to the licensee's previous proposal except as noted below.

The licensee's current IPC proposal differs from the previously approved case-specific IPC for D.C. Cook Unit No. 1 in several areas including:

1. the determination of the tube structural limit. Calculation of the tube structural limit has been based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions vice maintaining a margin of safety of 3 against burst during normal operation.

2. the IPC voltage limit. A 2.0-volt limit vice a 1-volt limit has been proposed.

3. the methodology for calculating postulated main steam line break (MSLB) leakage. Steam generator tube leakage during a postulated MSLB will be calculated in accordance with the methods described in draft NUREG-1477.

4. the diameter of the bobbin coil probe to be used in inspecting certain tubes. A 0.640-inch bobbin coil probe vice a 0.720-inch probe has been proposed for use in inspecting intersections which cannot be inspected with the 0.720-inch probe (e.g., intersections between sleeves).

To evaluate the 2.0-volt IPC proposal for D.C. Cook Unit No. 1, the staff considered not only the licensee's submittals but also operating experience from Farley Unit 2 (Southern Nuclear Operating Company letter dated January 19, 1994), foreign operating experience, and public comments received on draft NUREG-1477. The inservice inspection results from Farley Unit 2 (fall 1993) were used to assess the IPC methodology, since Farley Unit 2 was the first plant to operate a full cycle with a voltage-based IPC.

3.0 PROPOSED INTERIM PLUGGING CRITERIA

D.C. Cook Unit No. 1, TS 4.4.5 and 3.4.6.2 and Bases 3/4.4.5 and 3/4.4.6.2 are revised to specify the tube repair and leakage criteria for ODSCC at the tube support plate elevations for the fourteenth operating cycle. The tube repair and leakage criteria are based on the analysis in WCAP-13187, Revision 0, "D.C. Cook Unit 1 Steam Generator Tube Plugging Criteria for Indications at Tube Support Plates," and the analysis contained in the licensee's previously mentioned submittals. The changes to the tube repair and leakage criteria for cycle 14 are described below:

1. A 100% bobbin coil inspection of the hot leg tube support plate intersections and cold leg intersections down to the lowest cold leg tube support plate with known ODSCC indications will be performed.

2. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage less than or equal to 2.0 volts will be allowed to remain in service regardless of the depth of tube wall penetration, if the projected end-of-cycle leakage under postulated accident conditions (e.g., MSLB) will not result in the appropriate offsite dose limits being exceeded. The results of the MSLB leakage analysis will be reported to the NRC prior to restart for cycle 14.

3. Degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged except as noted in 4. below.

4. Indications of potential degradation within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to 3.6 volts may remain in service if an RPC probe inspection does not detect degradation. Indications of ODSCC degradation with a bobbin voltage greater than 3.6 volts will be plugged or repaired.

5. Primary-to-secondary leakage through all steam generators will be limited to 600 gallons per day (gpd) and primary-to-secondary leakage through any one steam generator will be limited to 150 gpd.

In addition to the above TS changes, the licensee also made the following proposals/commitments for implementing the IPC:

1. All flaw indications with bobbin voltages greater than 1.0 volt will be inspected using an RPC probe.

2. A sample RPC inspection of a minimum of 100 tube support plate intersections will be performed. All intersections with dent voltage exceeding 5.0 volts will be inspected by RPC. Inclusion of other intersections in the sample population will be based on inspecting intersections with artifact indications and intersections with unusual phase angles. Expansion of the sample plan, if required, will be based on the nature and number of the flaws discovered.

3. RPC flaw indications not found by the bobbin probe because of masking effects (due to denting, artifact indications, noise) will be plugged or repaired.

4. The NRC will be informed, prior to plant restart from the refueling outage, of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plate elevations. This includes any detectable circumferential indications or detectable indications outside the tube support plate.

5. The probability of tube burst during a postulated MSLB will be reported to the NRC prior to startup from the refueling outage.

4.0 EVALUATION

4.1 Tube Integrity Issues

The purpose of the TS 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 steam generator 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, limits on allowable primary-to-secondary leakage have been established in the TS to ensure timely plant shutdown before adequate structural and leakage integrity of the affected tube is impaired.

The proposed interim tube repair limits for D.C. Cook Unit No. 1 consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement.

The industry-wide database from the pulled tube examinations shows that for bobbin indications at or near 2.0 volts (i.e., the proposed IPC repair limit) maximum crack depths range between 50% and 100% 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 3.6 volts, the maximum crack depths have been found to generally range between 90% and 100% through-wall. Clearly, many of the tubes which will be allowed to remain in service under the proposed IPC may have or develop through-wall or near through-wall crack penetrations during the upcoming cycle, thus creating the potential for leakage during normal operation and postulated MSLB accidents. The NRC staff's evaluation of the proposed repair criteria from a structural and leakage integrity standpoint is provided in Sections 4.2 and 4.3, respectively. Section 4.4 contains the staff's evaluation of several inspection issues, and Section 4.5 addresses the importance of assessing the overall IPC methodology.

4.2 Structural Integrity

4.2.1 Burst Integrity

In support of the 1.0 volt repair limit approved in Reference 1 for the thirteenth operating cycle, the licensee developed a burst pressure/bobbin voltage correlation to demonstrate that bobbin indications satisfying the 1.0 volt interim repair criterion would retain adequate structural margins during cycle 13 operation, consistent with the criteria of Regulatory Guide 1.121. The correlation was developed from both pulled tube data (using pre-pull bobbin voltages) and laboratory tube specimens containing ODSCC flaws. The bobbin voltage data used to construct the burst pressure/bobbin voltage correlation were normalized to be consistent with the calibration standard voltage set-ups and voltage measurement procedures described in WCAP-13187 Revision 0 and most recently in the guidelines contained in the licensee's December 15, 1993, and February 15, 1994, submittals. The normalization was performed to ensure consistency among the voltage data in the burst pressure/bobbin voltage correlation and consistency between the voltage data in the correlation and the field voltage measurements at D.C. Cook Unit No. 1.

For D.C. Cook Unit No. 1, the most limiting burst pressure criterion of Regulatory Guide 1.121 is that degraded tubes shall retain a margin of 3 against burst under normal operating differential pressures across the tube. For D.C. Cook Unit No. 1, this translates to a limiting burst pressure of 4275 psi. From the most recent burst pressure/bobbin voltage correlation the maximum voltage which will satisfy this burst pressure criterion at a 95% prediction interval is approximately 4.9 volts. Since during normal operation the support plates provide constraint against tube rupture, the margin of safety of 3 against rupture during normal operation is inherently satisfied for flaws contained within the bounds of the tube support plates. Therefore, for ODSCC within the bounds of the tube support plates, the licensee has proposed that the tube structural limit should be based on maintaining a margin of safety of 1.43, consistent with Regulatory Guide 1.121, against tube failure under postulated accident conditions (e.g., MSLB) vice a factor of safety of 3 against burst during normal operation. For D.C. Cook Unit No. 1, this translates to a limiting burst pressure of 3660 psi. From the most recent burst pressure/bobbin voltage correlation the maximum voltage which will satisfy this burst pressure criterion at a 95% prediction interval is 9.6 volts.

In the licensee's December 15, 1993, and February 15, 1994 submittals, the 9.6 volt structural limit was adjusted to include an allowance of 20% for nondestructive examination (NDE) measurement uncertainty and an allowance of 40% for voltage growth over the next operating cycle to arrive at a 6.0 volt

APC repair limit. The NDE measurement uncertainty estimate considers measurement uncertainties stemming from bobbin coil probe design characteristics including wear characteristics and variability in the analysts' interpretation of the bobbin coil voltage. Potential flaw growth between inspections has been evaluated based on observed voltage amplitude changes during prior cycles at D.C. Cook Unit No. 1. Over the last three cycles, the average percent growth of all indications has been 42% (1987 to 1989), 42% (1989 to 1990), and 2.2% (1990 to 1992). The 40% average voltage growth allowance used to support the 6.0 volt APC repair limit is intended to provide margins for variation in future growth rates at D.C. Cook Unit No. 1. The licensee attributes the recent reduction in voltage growth, in part, to the reduction in the primary coolant hot leg temperature which was implemented in the 1989 time frame.

For any specific individual tube, voltage measurement uncertainty and/or voltage growth may exceed the value assumed in the previously mentioned Regulatory Guide 1.121 deterministic analysis since the deterministic analysis does not consider the full tails of the voltage measurement uncertainty and voltage growth distributions. Similarly, the burst pressure for some tubes may be less than the 95% lower prediction interval values in the burst pressure/bobbin voltage correlation. These distribution tails may involve sizable numbers of tubes in instances where a large number of tubes with indications are being accepted for continued service. To directly account for these uncertainties, Monte Carlo methods will be used to demonstrate that the probability of burst during a postulated MSLB accident 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 NDE uncertainty probability distribution and the voltage growth per cycle probability distribution. The EOC voltage distribution, the distribution of burst pressures, and a distribution of material tensile properties are then randomly sampled many times (e.g., 1,000,000) in order to determine the probability of burst during a postulated MSLB. In the probability of burst calculation, the material tensile properties distribution is sampled to adjust the burst pressure correlation which is based on a flow stress of 75 ksi. This probabilistic analysis allows for the possibility of burst pressures below those that were used to construct the burst pressure versus bobbin voltage correlation.

The licensee's submittals permit bobbin indications greater than 2.0 volts but less than 3.6 volts to remain in service if an RPC probe inspection does not detect a flaw, and it requires flaw indications with a bobbin voltage greater than 3.6 volts to be plugged or repaired. The staff notes that the 3.6 volts reflects an APC repair limit that was derived in WCAP-12871 Revision 2, "J.M. Farley Units 1 and 2, SG Tube Plugging Criteria for ODSCC at Tube Support Plates." In WCAP-12871 Revision 2, the APC repair limit was based on a structural limit of 3 times the normal operating pressure differential. The maximum voltage which would satisfy this burst pressure criterion at a 95% prediction interval was 6.2 volts based on the data available at that time. A 3.6 volt APC repair limit was calculated from the 6.2 volt structural limit by including an allowance of 20% for NDE measurement uncertainty and a 50% allowance for voltage growth over the next operating cycle. Since the issuance of WCAP-12871 Revision 2 in February 1992, additional data has been added to the burst pressure database used in the development of this APC voltage limit and several of the existing data points in the database have been updated as a result of additional analysis. In addition, it has been proposed that the voltage limit should be derived from a structural limit of 1.43xMSLB pressure vice 3 times the normal operating differential pressure as a result of the constraint provided by the tube support plate during normal operation. This has resulted in a new APC repair limit of 6.0 volts for D.C. Cook Unit No. 1.

To confirm the nature of the degradation occurring at the tube support plate elevations, the licensee has pulled several tubes from the steam generators during past outages. Tube pulls not only confirm the nature of the degradation but also provide data for assessing the reliability of the inspection methods and for supplementing existing databases (e.g., burst pressure, probability of leakage, and leak rate). Metallurgical examination performed on the tubes removed from D.C. Cook Unit No. 1 during the last refueling outage (i.e., 1992) confirmed that the dominant degradation mechanism for indications at the support plate elevations is axially oriented ODSCC. These examinations also revealed the presence of patches of axial and obliquely oriented cracks which formed a patch or cell-like structure. Corrosion within these patches was dominated by short, axial cracks, and the degradation was confined within the support plate crevice region. The maximum voltage of the intersections removed was 2.02 volts, and the burst pressures for the specimens ranged from 9,100 psi to 11,200 psi. The staff believes that no additional pulled tube data is required to support implementation of the 2.0 volt IPC during the 1994 refueling outage provided no unusual inspection findings (described previously) are found during the inspection.

The staff concludes that the proposed 2.0 volt interim criterion will provide adequate assurance that most tubes with indications which are accepted for continued service during cycle 14 operation will meet the burst pressure criteria of Regulatory Guide 1.121 at the end of cycle 14. The staff notes that the bounding value of voltage growth per cycle at D.C. Cook Unit No. 1 since 1987 has not exceeded 0.8 volt (the maximum growth during the last cycle (Cycle 12) was 0.49 volt). The staff estimates the 0.8 volt to represent a bounding value, assuming no increase in corrosion rates over what has been observed previously at D.C. Cook Unit No. 1. Assuming a 20% voltage measurement uncertainty for a 2.0 volt indication left in service, the EOC voltage is expected by the staff to be bounded by 3.2 volts. This is below the 9.6 volt structural limit evaluated by the licensee as the lower 95% confidence limit for meeting the burst pressure criterion of 1.43xMSLB pressure using the most recent burst pressure correlation. The staff also concludes that for axially oriented ODSCC within the bounds of the tube support plates, a structural limit based on maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions (e.g., MSLB) is acceptable since tube support plate constraint during normal operation inherently satisfies the margin of safety of 3 during normal operation.

The staff also concludes that the proposal to allow bobbin indications between 2.0 and 3.6 volts to remain in service provided that the RPC probe inspection does not confirm the degradation observed with the bobbin coil probe is acceptable. The staff notes that short and/or relatively shallow cracks detected by the bobbin coil may sometimes not be detectable by the RPC probe, although the RPC probe 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/bobbin voltage correlation. The staff believes that the burst pressure for bobbin indications which were not confirmed to be flaw-like by the RPC probe will tend to be at the upper end of the burst pressure distribution (i.e., exhibit a higher burst pressure). The 3.6 volt cutoff, such that all bobbin indications would be plugged or repaired (with or without confirming RPC indications), provides assurance that all excessively degraded tubes will be removed from service. The staff notes that, even if a 3.6 volt indication were left in service, assuming an allowance of 20% for measurement uncertainty and a growth rate of 1.6 volts (i.e., twice the maximum growth rate observed since 1987), the EOC voltage would be 5.9 volts. This 5.9 volts provides significant margin relative to the 9.6 volt structural limit. The staff further notes that the projected leakage from these tubes (i.e., tubes with bobbin voltages between 2.0 and 3.6 volts which exhibited no detectable degradation during the RPC inspection) will be considered in the leak rate assessment performed by the licensee prior to plant restart. Thus, the staff finds the proposed exception to the 2.0 volt criterion to be acceptable.

Furthermore, the staff concludes that the methodology for calculating the conditional probability of burst given an MSLB, referenced above, should use a BOC distribution that includes: (1) all indications including those that were not confirmed by the RPC probe to be degraded, and (2) non-detected ODSCC indications. In addition, the burst pressure correlation used in these calculations should include all data unless a specific error in either the burst pressure test or voltage measurement occurred. The licensee has committed to perform such an analysis following the inspection at D.C. Cook Unit No. 1 to confirm an acceptably low probability of burst given a MSLB. The results of this analysis (which should consider the most recent burst pressure/bobbin voltage correlation and the most recent growth rate data) should be reported to the staff prior to plant startup from the refueling outage. The staff notes a calculation similar to this (excluding the corrections for the RPC non-confirmed indications and non-detected ODSCC indications) was performed following implementation of a 1.0 volt IPC at D.C. Cook Unit No. 1 which indicated that implementation of a 1.0 volt repair criterion at that time would have yielded a conditional probability of burst given an MSLB of zero (i.e., no occurrences in 100,000 Monte Carlo samples).

4.2.2 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 MSLB loads on tube integrity, consistent with 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 tube support plates 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 leakage) under LOCA conditions. In-leakage during a 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 for the combination of SSE and LOCA loads. The seismic excitation defined for steam generators is in the form of acceleration response spectra at the steam generator support. 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 51 steam generator 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 tube support plate 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 D.C. Cook Unit No. 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-vibrations 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 tube support plates.

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 steam generator caused by the break hydraulics and reactor coolant loop motion. However, the resulting tube support plate 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 steam generator, the reactor coolant pump, and the primary piping. This analysis yields the time-history displacements of the steam generator 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 steam generator.

In calculating a combined tube support plate 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 tube support plate load was transferred to the steam generator 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 tube support plate 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 D.C. Cook Unit No. 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 D.C. Cook Unit No. 1, no significant tube deformation or leakage is likely to occur during an SSE plus LOCA event. In addition, the burst strength of tubing with through-wall cracks is not affected by an SSE event.

4.3 Leakage Integrity

A number of the indications satisfying the proposed interim 2.0 volt repair limit can be expected to have, or to develop, through-wall 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 TS limits on allowable primary-to-secondary leakage. Adequate leakage integrity during transients and postulated accidents is demonstrated by showing that for the most limiting accident, assumed to occur at the end of the next cycle, the resulting leakage will not exceed a rate that will result in offsite dose limits being exceeded.

4.3.1 Normal Operating Leakage

The licensee has proposed an interim change to the reactor coolant system leakage limit criteria in TS 3.4.6.2 that is applicable to the fourteenth operating cycle. Specifically, the licensee has proposed to continue to limit the primary-to-secondary leakage through any one steam generator to 150 gpd and to continue to limit the total primary-to-secondary leakage through all steam generators to 600 gpd. The standard primary-to-secondary leakage limits for D.C. Cook Unit No. 1 are 500 gpd through any one steam generator and 1.0 gpm (1440 gpd) through all steam generators.

The standard 500 gpd limit per steam generator 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. Development of the 150 gpd per steam generator interim leakage limit has utilized the extensive industry database 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 150 gpd limit would be exceeded before the crack length reaches the critical crack length for MSLB pressures. Based on nominal, best estimate leakage rates, the 150 gpd limit would be exceeded before the crack length reaches the critical crack length corresponding to a burst pressure of 3 times normal operating pressure.

The interim leakage limits are more restrictive than the standard operating leakage limits in order to provide a margin of safety against rupture. The interim leakage 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 tube support plate, and thus provide additional protection against exceeding MSLB leakage limits. The staff finds the proposed interim operating leakage limits in TS 3.4.6.2 to be acceptable for implementation of the IPC.

4.3.2 Accident Leakage

As the basis for estimating the potential leakage during MSLB accidents, Westinghouse has correlated leakage test data obtained under simulated MSLB conditions with the corresponding bobbin voltage amplitudes. The correlation is based on a linear regression fit of the logarithms of the corresponding leak rates and bobbin voltages. The leak rate data exhibits considerable scatter relative to the mean correlation. Thus, prediction intervals for leak rate at a given voltage have been established to statistically define the range of potential leak rates. As part of the on-going review of the APC, the staff is continuing to review the correlation of the leak rate data to bobbin voltage. The staff tentatively concluded in draft NUREG-1477 that no proven relationship between leakage rate and voltage presently exists and that the proposed approach fails to account for non-detected ODSCC that remains in service. The staff has also noted that there are very few leakage data points in the 0-to-3 volt range.

However, until the issue of the leak rate versus voltage correlation is resolved, the staff has concluded that a voltage-based approach can be used if these nonconservatisms are accounted for and sufficient conservatisms are included in the analysis. Therefore, the licensee has incorporated a requirement in the proposed TS to provide a calculation of potential MSLB leakage by a methodology designed to address the staff concerns. The methodology that the licensee will use to calculate the MSLB leakage is described in draft NUREG-1477. This methodology treats the leakage rate data as independent of voltage. The staff notes that the MSLB leakage analysis should be performed with the most recent leak rate data for 7/8-inch outside diameter tubing. In addition, the voltage growth distribution used in the leakage assessment should (1) consider the most recent voltage growth data (i.e., cycle 13), and (2) be adjusted for the planned cycle 14 duration. Evaluation of the acceptability of the estimated primary-to-secondary leakage rate for postulated accident conditions should be based on current staff positions regarding calculational methods and limits for offsite doses.

The staff noted in draft NUREG-1477 that there was no theoretical basis for assuming a log logistic fit for the probability of leakage function. Furthermore, the staff noted that the form of fit could significantly affect the predicted leakage but that the results would vary depending on the EOC voltage distribution. The staff concludes, therefore, that for the 2.0 volt IPC at D.C. Cook Unit No. 1, the most conservative (with respect to the overall leakage) of the six functional forms for the probability of leakage function (discussed in draft NUREG-1477) should be used in predicting the primary-to-secondary leakage during a postulated MSLB.

The staff also notes that the postulated accident leakage methodology described in draft NUREG-1477 does not account for the potential uncertainties in the leakage parameters (e.g., uncertainties in the mean and standard deviation of the leak rate data). However, the staff believes that even if these uncertainties were accounted for in the D.C. Cook Unit No. 1 analysis that the predicted leakage would be acceptable with respect to offsite dose considerations. This conclusion is based, in part, on the previous inspection results which indicated minimal tube degradation at D.C. Cook Unit No. 1. The staff, therefore, concludes that these parametric uncertainties need not be included in the leakage analysis for this cycle.

4.4 Inspection Issues

In support of the proposed interim repair limit, the licensee proposes to utilize the eddy current test guidelines provided in Attachment 6 to its December 15, 1993, submittal, as supplemented by letters dated February 15, 1994, and February 24, 1994, to ensure the field bobbin indication voltage measurements are obtained in a manner consistent with the development and analyses of the supporting databases. The proposed guidelines define, in part, the bobbin specifications, calibration requirements, specific acquisition and analyses criteria, and flaw recording guidelines to be used for the inspection of the steam generators.

The proposed inspection guidelines and other licensee commitments contain, in part, requirements to (1) record all indications regardless of voltage amplitude (required for assessing postulated MSLB leakage and probability of burst), (2) perform RPC inspections of 100 tubes, including all tubes with bobbin dent voltages exceeding 5 volts and also including tube support plate intersections with artifact indications or indications with unusual phase angles (expansion of this sample, if required, will be based on the nature and number of the flaws discovered), (3) perform RPC examinations of all tubes with bobbin voltages in excess of 1.0 volt, and (4) inform the staff prior to plant restart from the refueling outage of any unexpected RPC findings relative to the assumed characteristics of the flaws at the tube support plate elevations (which includes any detectable circumferential indications or detectable indications extending outside the thickness of the tube support plate).

The staff notes that the proposed NDE guidelines contain modifications to the previously used guidelines. These modifications include, in part, a discussion on the adequacy of RPC probes (i.e., 1-, 2-, or 3-coil) at distinguishing crack characteristics and provisions to reduce the tube repair limit for tubes inspected with a probe where the probe wear limit was exceeded. The staff is reviewing these changes with respect to the generic implementation of a 2.0 volt repair criterion; however, the staff concludes that these modifications are acceptable for the 1994 refueling outage at D.C. Cook Unit No. 1.

The staff notes that the original calibration procedure for the bobbin coil presented in earlier APC submittals, which requires setting the bobbin coil voltage amplitude from the 400/100 kHz differential channel from the four 100% through-wall holes, is preferred over the more recent guidelines which require calibration on the four 20% through-wall holes, as discussed in draft NUREG-1477. In addition, the staff notes that there are several outstanding technical issues pertaining to the inspection guidelines, as documented in draft NUREG-1477, that will require resolution prior to adopting higher voltage limits.

As part of this IPC proposal, the licensee has proposed to use smaller diameter bobbin probes to inspect intersections which cannot be accessed using the standard 0.720" bobbin probe (i.e., intersections between sleeved locations). To support the use of a smaller diameter bobbin probe, the licensee provided results from two plants, where a limited number of tubes were inspected with both the standard 0.720-inch bobbin probe and a smaller diameter bobbin probe (i.e., 0.560-inch, 0.580-inch, and 0.640-inch bobbin probes). The results from these tests demonstrated that the voltages measured with the smaller diameter bobbin probe were equal to or greater than the voltages measured with the larger diameter bobbin probe for the majority of the indications. However, the analysis provided was limited and did not account for the potentially higher noise levels on the detectability of the flaws when using smaller diameter probes. The staff concludes, therefore, that the use of smaller diameter bobbin probes is acceptable only if the licensee performs a more rigorous statistical analysis to demonstrate the adequacy of the smaller diameter bobbin probes not only to size but also to detect the indications. The analysis methodology for performing such a demonstration should be submitted for NRC review and approval. As a result of the staff concerns, the licensee proposed in letters dated February 15, 1994, and February 24, 1994, that:

1. if the smaller diameter probe is used in conjunction with the IPC, it will be the subject of separate correspondence with the NRC, and

2. if this statistical analysis is not conducted, tube repairs for tubes requiring inspection with a smaller diameter probe will be based on the standard TS criterion of 40% through-wall as determined with the smaller diameter bobbin probe.

The staff finds this proposal acceptable for D.C. Cook Unit No. 1; however, the statistical analysis methodology must be approved by the NRC staff prior to implementing IPC repairs on tubes inspected with the smaller diameter bobbin probes. The staff is also evaluating the generic aspects of using smaller diameter bobbin probes.

4.5 Overall Assessment of IPC Methodology

Draft NUREG-1477, issued by the NRC in June 1993, provided the conclusion of an NRC task force with regard to a 1.0 volt tube repair criteria. In that report the staff noted that there is not a unique relationship between eddy current voltage amplitude and crack depth and length and that this lack of a unique relationship is reflected in the scatter of the tube burst pressure and leakage data when plotted as a function of voltage. In this regard, the task group concluded that a voltage-based approach can be used if appropriate conservatisms are included in the statistical analysis. The staff has considered this conclusion in its current evaluation and has determined that adequate margin exist with regard to assumed burst pressure behavior, degradation rates, NDE variability, and leakage calculation to support this plant-specific implementation of a 2.0/3.6 volt IPC. The staff is continuing its evaluation of the public comments received on draft NUREG-1477 and notes that resolution of several outstanding technical issues (e.g., handling of outliers, limited pulled tube database above 3.6 volts, NDE uncertainty model, voltage growth model, need for additional operating experience, etc) will be necessary to implement higher voltage limits. Several of the staff positions are supported by the most recent operating experience data (e.g., probability of detection adjustment to account for new indications, performance demonstration to reduce analyst variability, etc.) from Farley Unit 2. The staff has concluded, however, that the 2.0/3.6 volt IPC is acceptable (as documented above) to ensure tube structural integrity for this plant-specific application.

The licensee has committed to perform an assessment of the effectiveness of the methodology described in WCAP-13187 Revision 0 for predicting the EOC voltage distribution. The assessment will address any discrepancies between the predicted and actual values. The following information will be included in this assessment:

1. EOC 12 voltage distribution - indications found during the inspection regardless of RPC verification

2. Cycle 12 growth rate (i.e., from BOC 12 to EOC 12)

3. EOC 12 repaired indications voltage distribution - distribution of indications presented in 1. above that were repaired (i.e., plugged or sleeved)

4. Voltage distribution for indications left in service at the BOC 13 regardless of RPC confirmation - obtained from 1. and 3., above

5. Voltage distribution for indications left in service at the BOC 13 that were confirmed by RPC to be crack-like or not RPC inspected

6. Non-destructive examination uncertainty distribution used in predicting the EOC 13 voltage distribution

7. Projected EOC 13 voltage distribution

8. Actual EOC 13 voltage distribution - all indications found during the inspection regardless of RPC confirmation

9. Cycle 13 growth rate (i.e., from BOC 13 to EOC 13)

10. EOC 13 repaired indications voltage distribution - distribution of indications presented in 8. above that were repaired (i.e., plugged or sleeved)

11. Voltage distribution for indications left in service at the BOC 14 regardless of RPC confirmation - obtained from 8.(h) and 10.(j) above

12. Voltage distribution for indications left in service at the BOC 14 that were confirmed by RPC to be crack-like or not RPC inspected

13. Non-destructive examination uncertainty distribution used in predicting the EOC 14 voltage distribution

14. Projected EOC 14 voltage distribution

The licensee has committed to submit this assessment approximately 10 weeks from completion of the steam generator inspections. The staff finds this acceptable; however, it requests that the above information be submitted in both tabular and graphical form.

4.6 Radiological Consequences

The base analysis for the licensee's proposal was provided in their submittal dated February 15, 1994. This analysis determined the maximum permissible steam generator (SG) primary-to-secondary leak rate during a main steam line break (MSLB) for Cook Unit 1 considering the accident initiated iodine spike case. The licensee, in performing its analyses, considered the acceptance criteria of SRP Sections 15.1.5 Appendix A. As a result of their analysis, the licensee concluded that the limiting primary-to-secondary MSLB leakage would be limited to 12.6 gpm in the faulted SG so that accident consequences remain within SRP acceptance criteria. The staff has reviewed the licensee's analysis and performed an independent analysis of the radiological consequences of a MSLB w/primary-to-secondary leakage of 12.6 gpm for both the event generated spike case and the pre-existing iodine spike case and has determined that the acceptance criteria of SRP Sections 15.1.5 Appendix A are satisfied.

The calculated MSLB leakage as determined by the licensee using the methodology discussed in section 4.3.2 of this safety evaluation must be below the proposed leakage limits, or additional tubes must be repaired until the leakage is within the limits.

4.7 Severe Accident Impact

Draft NUREG 1477 (Section 4.4) addressed severe accident analysis with respect to steam generator tube IPC. The staff accepted IPC intending to maintain the current level of steam generator tube integrity, consistent with Regulatory Guide 1.121. This approach was considered credible since the degradation mechanism addressed is confined to regions within the tube support plate. The staff judged that expected tube performance would not be significantly impacted, so that high pressure severe accident analyses would not be affected.

The application proposing a revised IPC addresses analyses to demonstrate adequate tube structural and leakage integrity. These analyses ensure that the tube structural integrity guidelines of Regulatory Guide 1.121 are met. As detailed elsewhere in this evaluation, extending IPC to include higher voltage indications does not significantly alter accepted tube integrity. Therefore, the tube behavior for normal operation or transients is not expected to be markedly degraded. It is the staff's judgment that the effect on high pressure severe accident response of this change is within the uncertainties associated with severe accident analysis capabilities. Therefore, the basis for the staff conclusion reported in NUREG-1477 regarding severe accident impact is unchanged. That is, the staff judges that under a higher voltage IPC, expected tube performance would not be impacted sufficiently to alter high pressure severe accident analyses.

4.8 Summary

Based on the above evaluation, it can be concluded that adequate structural integrity of the steam generator tubing can be ensured for cycle 14 at D.C. Cook Unit No. 1, consistent with applicable regulatory requirements. In addition, the staff concludes that proposed interim operating leakage limits and the methodology described above for determining the expected primary-to-secondary leakage during a postulated MSLB at the end of fuel cycle 14 for D.C. Cook Unit No. 1 is acceptable. The staff's approval of the proposed interim repair limit is based on the licensee being able to demonstrate that the primary-to-secondary leakage during a postulated MSLB will be acceptable. The licensee has agreed to report, prior to plant startup from the refueling outage, the results of the MSLB leakage analysis. The licensee has also agreed to inform the NRC prior to plant startup from the refueling outage of any unexpected inspection findings relative to the assumed characteristics of the flaws at the tube support plates. This includes any detectable circumferential indications or detectable indications outside the tube support plate thickness.

5.0 EXIGENT CIRCUMSTANCES

Although a Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for Hearing was published in the Federal Register (January 5, 1994, 59 FR 621) for the December 15, 1993, application, new information beyond the scope of the December 1993 application was submitted by the licensee by letter dated February 15, 1994. In its February 15, 1994 letter, the licensee requested that its application for the license amendment be processed as involving exigent circumstances.

The Commission's regulation, 10 CFR 50.91, provides special exceptions for the issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case in which the staff and the licensee need to act quickly and time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and the Commission also determines that the amendment involves no significant hazards considerations. In this instance, D.C. Cook, Unit No. 1, would have to plug a greater number of steam generator tubes following ECT thereby reducing the heat transfer capability from the primary to secondary system. Such a reduction would have a safety and economic impact. In accordance with 10 CFR 50.91(a)(6)(i)(B), the Commission used local media to provide reasonable notice to the public in the area surrounding the D.C. Cook Nuclear Power Plant facility of the licensee's proposed amendment and of the Commission's proposed determination that a no significant hazards consideration is involved.

The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on March 7, 1994. The notice was published in the South Haven Tribune on March 1, 1994, and in the Herald-Palladium on March 2, 1994.

The exigent circumstances resulted from a recent change in NRC staff acceptance of higher interim voltage limits (i.e., 2.0 volts). The change was made known to the licensee during a meeting on February 9, 1994. As a result of this meeting, the licensee requested a license amendment to incorporate the 2.0 volt criterion into the Unit No. 1 steam generator inservice inspection and repair program during the current Unit No. 1 refueling outage.

The staff finds that the licensee did not deliberately or negligently cause the exigent situation to come into being. Failure of the Commission to act on the licensee's request would result in additional plugging of steam generator tubes.

6.0 FINAL NO SIGNIFICANT HAZARDS DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee has analyzed the proposed amendment to determine if a significant hazard consideration exists:

> (1) Operation of the Donald C. Cook Nuclear Plant Unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.
>
> Testing of model boiler specimens for free span tubing (no tube support plate restraint) at room temperature conditions show burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on pulled tubes from Cook Nuclear Plant Unit 1 with up to a 2.02 volt indication shows measured burst pressure in excess of 10,000 psi a room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5 volt indications show burst pressures in excess of 6,300 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety factor requirements of Regulatory Guide (RG) 1.121. As stated earlier [in the application], tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate. Test data indicates that tube burst cannot occur within the tube support plate, even for tubes which have 100% throughwall electric-discharge machined (EDM) notches 0.75 inch long, provided that the tube support plate is adjacent to the notched area. Since tube to tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintains the RG 1.121 margin of safety of 1.43 times the bounding faulted condition (steamline break) pressure differential.
>
> During a postulated main steamline break, the TSP [tube support plate] has the potential to deflect during blowdown, thereby uncovering the intersection. Based on the existing data base, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the steamline break pressure differential on tube burst is satisfied by 7/8 inch diameter tubing with bobbin coil indications with signal amplitudes less than 9.6 volts, regardless of the indicated depth measurement. A 2.0 volt plugging criteria compares favorably with the 9.6 volt structural limit considering the previously calculated growth rates for ODSCC [outer diameter stress corrosion cracking] within the Cook Nuclear Plant Unit 1 steam generators.

Considering a voltage growth component of 0.8 volts (40% voltage growth based on 2.0 volts BOC and an NDE [nondestructive examination] uncertainty of 0.40 volts (20% voltage uncertainty based on 2.0 volts BOC), when added to the BOC interim plugging criteria of 2.0 volts results in a bounding EOC voltage of approximately 3.2 volts for Cycle 14 operation. A 6.4 volt safety margin exists (9.6 structural limit - 3.2 volt EOC = 6.4 volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 9.6 volts. Using this structural limit of 9.6 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit the existence of EOC indications which exceed the 9.6 volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Previous IPC submittals have established the conservative NDE uncertainty limit of 20% of the BOC repair limit. For consistency, a 40% voltage growth allowance to the BOC repair limit is also included. This allowance is extremely conservative for Cook Nuclear Plant Unit 1. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 9.6 volts can be represented by the expression:

$$RL + (0.2 \times RL) + (0.4 \times RL) = 9.6 \text{ volts, or,}$$

the maximum allowable BOC repair limit (RL) can be expressed as,

$$RL = 9.6 \text{ volt structural limit}/1.6 = 6.0 \text{ volts.}$$

This structural repair limit supports this application for cycle 14 IPC implementation to repair bobbin indications greater than 2.0 volts independent of RPC [rotating pancake coil] confirmation of the indication. Conservatively, an upper limit of 3.6 volts will be used to assess tube integrity for those bobbin indications which are above 2.0 volts but do not have confirming RPC calls.

The conservatism of this repair limit is shown by the EOC 12 (summer 1992) eddy current data. The overall average voltage growth was determined to be only 2.2% (of the BOC voltage), with a 12% average voltage growth for indications less than 1.0 volt BOC and a 1% average voltage growth for indications greater than 0.75 volts at the BOC. In addition, the EOC 12 maximum observed voltage increase was found to be 0.49 volts, and occurred in a tube initially less than 1.0 volt BOC. The applicability of cycle 13 growth rates for cycle 14 operation will be confirmed prior to return to service of Cook Nuclear Plant Unit 1. Similar large structural margins are anticipated.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main steamline break outside of containment but upstream of the main steam isolation valve represents the most limiting radiological condition relative to the IPC. In support of implementation of the IPC it will be determined whether the distribution of crack indications at the tube support plate intersections at the end of cycle 14 are projected to be such that primary to secondary leakage would result in site boundary doses within a small fraction of the 10 CFR Part 100 guidelines. A separate calculation has determined this allowable steamline break leakage limit to be 12.6 gpm. Although not required by the Cook Nuclear Plant design basis, this calculation uses the recommended Iodine-131 transient spiking values consistent with NUREG-0800, and the TS reactor coolant system activity limit of 1.0 micro curie per gram dose equivalent Iodine-131. The projected steamline break leakage rate calculation methodology prescribed in Section 3.3 of draft NUREG-1477 will be used to calculate EOC 14 leakage. Due to the relatively low voltage growth rates at Cook Nuclear Plant Unit 1 and the relatively small number of indications affected by the IPC, steamline break leakage prediction per draft NUREG-1477 is expected to be less than the acceptance limit of 12.6 gpm in the faulted loop.

Application of the criteria requires the projection of postulated steamline break leakage, based on the EOC voltage distribution. EOC voltage distribution is developed using EOC-13 eddy current results and a voltage measurement uncertainty. The data indicates that a threshold voltage of 2.81 volts would result in throughwall cracks long enough to leak at steamline break conditions. Draft NUREG-1477 requires that all indications to which the IPC are applied must be included in the leakage projection. Tube pull results from Cook Nuclear Plant Unit 1 indicate that tube wall degradation of greater than 40% throughwall was detectable either by the bobbin or RPC probe. The tube with maximum throughwall penetration of 56% (43% average) had a voltage of 2.02 volts. This indication also was the largest recorded bobbin voltage from the EOC 12 leakage of 2.81 volts, inclusion of all IPC intersections in the leakage model is quite conservative. Therefore, as implementation of the 2.0 volt IPC during cycle 14 does not adversely affect steam generator tube integrity and results in acceptable dose consequences, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the Cook Nuclear Plant Unit 1 FSAR [Final Safety Analysis Report].

(2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed steam generator tube IPC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the tube support plate elevations; no ODSCC is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied (during all plant conditions).

Specifically, we will continue to implement a maximum leakage rate limit of 150 gpd (0.1 gpm) per steam generator to help preclude the potential for excessive leakage during all plant conditions. The cycle 14 TS limits on primary to secondary leakage at operating conditions is a maximum of 0.4 gpm (600 gpd) for all steam generators, or, a maximum of 150 gpd for any one steam generator.

The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 9.6 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95% prediction limit on the burst correlation coupled with 95/95 lower tolerance limit (LTL) material properties. Alternate crack morphologies can correspond to 9.6 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 1.43 times the steamline break pressure differential (1.43 x 2560 psi = 3660 psi) and the steamline break pressure differential alone (2560 psi) are approximately 0.53 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.42 inch long cracks at nominal leak rates and 0.61 inch long cracks at the lower 95% confidence level leak rates. Since tube burst is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during steamline break conditions, the leakage from the maximum permissible crack must preclude tube burst at steamline break conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steamline break conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncovery will provide benefit to the burst capacity of the intersection.

(3) The proposed license amendment does not involve a significant reduction in margin of safety.

The use of the voltage based bobbin probe interim tube support plate elevation plugging criteria at Cook Nuclear Plant Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC 14 distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of a loss of coolant accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow areas increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the Cook Nuclear Plant Unit 1 reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the Cook Nuclear Plant Unit 1 smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volts is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the tube support plate elevations per TS, and RPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the tube support plate elevation plugging criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the interim plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the FSAR or any Bases of the plant TS.

Based on the above considerations, including the staff's safety evaluation, the staff concludes that the amendment meets the standards set forth in 10 CFR 50.92 for a no significant hazards determination. Therefore, the staff has made a final determination that the proposed amendment involves no significant hazards consideration.

7.0 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.

8.0 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 surveillance requirements. The 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 (59 FR 621). Further, the proposed finding for the supplemental information was issued in the local media described in Section 4.0 of this safety evaluation. 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.

9.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that because the requested changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration; and 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.

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.

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DATED: March 15, 1994

AMENDMENT NO.178 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK UNIT NO.1

Docket File
NRC & Local PDRs
PDIII-1 Reading
J. Roe
J. Zwolinski
L. Marsh
J. Hickman
C. Jamerson
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
R. Jones
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