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TECHNICAL EVALUATION REPORT
OF
"STEAM GENERATOR REPAIR REPORT"

DONALD C. COOK NUCLEAR PLANT
UNIT NO. 2

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1. INTRODUCTION

On November 7, 1986, Indiana and Michigan Electric Corporation submitted a report entitled "Steam Generator Repair Report" that describes the safety-related aspects associated with the repair of the D.C. Cook Unit 2 steam generators. This involves the replacement of the existing units with shop fabricated lower assemblies.

The D.C. Cook report has been reviewed and evaluated to insure that the unit can be repaired and operated without undue risk to public health and safety. This evaluation focuses on changes in steam generator mechanical design, thermal hydraulics, materials selection, fabrication techniques and changes in secondary systems design and operation intended to minimize the potential for corrosion and steam generator tube degradation encountered in the original steam generators.

2. BACKGROUND

D.C. Cook Unit 2 has been operating since March 10, 1978. It has four Westinghouse Series 51 steam generators, each of which has 3388 tubes made from Inconel 600. These mill annealed tubes are hardrolled for a distance of approximately 2.25 inches above the bottom of the tubesheet, leaving an annular crevice of approximately 18.75 inches with a 0.007 to 0.009 inch radial gap. Tube support plates are carbon steel with drilled tube holes having a nominal radial clearance of 0.008 inches.

D.C. Cook Unit 2 has been using all volatile treatment (AVT) with hydrazine and ammonia for secondary water treatment throughout its operating life. Until 1983 only minor tube degradation involving wear at antivibration bar (AVB) intersections was noted during inspections. However, row 1 U-bend tubes experienced primary side cracking and all row 1 tubes were preventively plugged in 1984.

By early 1984 leakage resulting from secondary side corrosion was occurring with increasing frequency. Eddy current testing (ECT) revealed degradation just above the tubesheet and in the tubesheet crevice region. Representative tubes were removed and metallographic examination revealed intergranular attack/stress corrosion cracking (IGA/SCC), probably caused by concentrated caustic material in the sludge pile and in the tubesheet crevices. Boric acid soaks were initiated to neutralize the alkaline environment in an attempt to slow the rate of IGA/SCC.

A series of leaks, post-leak inspections, and tube plugging continued through 1986. Eddy current tests and metallurgical examinations of removed tubes revealed attack in the tubesheet area and the presence of axially oriented intergranular stress corrosion cracks (IGSCC) in the tubes at tube support plate locations.

As a result of the continued corrosion-related degradation of the steam generator tubes, D.C. Cook Unit 2 has frequently examined and plugged steam generator tubes to ensure continued plant operation. In addition, D.C. Cook Unit 2 initiated on-line boric acid treatment and reduction in operating temperature (and power) in order to arrest the rate of steam generator tube degradation. At the end of 1986 there were 743 plugged tubes in the D.C. Cook Unit 2 steam generators. A continuation of the tube plugging trend indicates that the Technical Specification plugging limit could be reached in the near future and result in permanent power limitations. Therefore, to increase availability and to return to full power operation, Indiana and Michigan Electric Power Co. plans to repair the four D.C. Cook Unit 2 steam generators by replacing the lower assemblies.

3. DISCUSSION

In the repair report Indiana and Michigan Electric Power Co. describes its repair plans as follows:

The D.C. Cook Unit 2 steam generator repair project will involve the reactor coolant system pipe-cutting process to remove the steam generator lower assemblies. This technique was used successfully in the Surry Units and the Point Beach Unit 1 steam generator replacement programs. Using this method, each steam generator will be cut just above the existing transition cone girth weld in the upper shell and at the inlet and outlet reactor coolant piping nozzles. The upper assembly will be removed from containment through the equipment hatch for modifications to the moisture separation and feedring equipment. Both the original and replacement steam generator lower assemblies will also be taken out of and into containment through the equipment hatch. The original upper assembly, the new lower assembly, and the associated piping will be welded together in the field.

The repair program in its entirety will be done in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition and Addenda through the Summer of 1983. The licensee also states that the original design configurations and installation requirements will be used when components and piping are reinstalled. This will eliminate any design modifications that would require changes to the original design analysis.

Westinghouse will fabricate the replacement steam generator lower assemblies. The design of the replacement lower assemblies is purported to match the design performance of the original lower assemblies. It includes some new features that will not alter mechanical performance. The new design features are intended to provide improved thermal hydraulic performance, minimize secondary side corrosion, and facilitate access to the tube bundle. A discussion of the replacement steam generator design as presented by Indiana and Michigan Electric Power Co. is as follows:

3.1 Replacement Steam Generator Design

The replaced steam generators will be similar in design and will be functionally the same as the original steam generators. Design changes have been made to address the operating experience difficulties that led to the replacement of the original steam generators. The primary objectives of the design changes are to increase resistance to known degradation mechanisms and improve overall performance while enhancing reliability and maintainability of the steam generators.

Major changes will involve refurbishing the upper assembly with a new feedring with Inconel 600 J-nozzles and upgrading the separators to enhance performance. This refurbishment of the upper assembly will be done in the field.

The steam generator tubes will be made of thermally treated Inconel 690. Increased heat transfer area will be provided by an increase in the number of tubes, achieved by reducing the pitch of the tubes. To minimize the tubesheet crevice the tubes will be hydraulically expanded along the entire length in the tubesheet. The inside eight rows of tubes will be heat-treated to relieve the residual stresses resulting from the bending operation.

The tube support plates will be made of stainless steel with quatrefoil tube holes. A flow distribution baffle and tube lane flow blocker are also provided in the replacement design. Three sets of antivibration bars will be installed to support the tubes in the U-bend region.

Additional features designed to enhance maintainability include six handholes: four below the flow distribution baffle but above the tubesheet and two located above the flow distribution baffle. The grid location of the tubes will be permanently marked on the primary side tube sheet.

3.1.1 Design Improvement to Overcome Tube Degradation

Design improvements to overcome tube degradation address the issue of intergranular corrosion (IGC), primary side SCC and wear at AVB locations.

3.1.1.1 Intergranular Corrosion

A pervasive problem with the original D.C. Cook Unit 2 steam generators was the poor resistance of the mill annealed Inconel 600 tubing to IGC in general, and particularly in the tubesheet region and at tube support plate locations. The tube-to-tubesheet and tube-to-support plate crevices provided a localized aggressive corrosive environment because of the concentration of impurities from the feedwater. Additionally, corrosion products from the carbon steel support plates filled the gap and squeezed the tube, adding stress and contributing to IGC at the tube support location. Thermal hydraulic characteristics resulted in dry-out and sludge accumulation, which provided another localized aggressive environment for IGC of the Inconel 600 tubing just above the tubesheet surface.

Indiana and Michigan Electric Power Co. contends that by using tubing made of thermally treated Inconel 690 (I-690 TT) in lieu of mill annealed Inconel 600 (I-600 MA) the degree of corrosion-initiated tube degradation would be reduced. The utility anticipates a reduction because I-690 TT is

reported to have high corrosion resistance because of the heat treated microstructure.

The combination of corrosion resistant tube material and full depth tube expansion in the tubesheet, to eliminate the tubesheet crevice and site for impurity concentration, is expected to reduce IGC in the tubesheet area.

Intergranular corrosion at tube support locations is also expected to be minimized by the use of ferritic stainless steel support plates with a quatrefoil hole design. The quatrefoil hole design has a lower pressure drop and a high sweeping flow and velocity around the tubes, which reduces the potential for buildup of contaminant at support locations. The combination of high velocities and flow and the corrosion resistance of the stainless steel plate material reduces the potential for tube IGC at the support plate locations.

Corrosion attack of tubes exposed to sludge build-up on the tubesheet is expected to be reduced by the use of a flow distribution baffle of ferritic stainless steel located approximately 23 inches above the tubesheet. The flow distribution baffle will have octafoil broached tube holes surrounding an open central cutout. The baffle directs flow radially across the tubesheet and up the center of the bundle through the central cutout, increasing lateral flow velocities across the tubesheet. This results in the deposition of sludge near the center of the bundle at the blowdown intake, reducing the number of tubes exposed to the sludge. Additionally, the octafoil hole design reduces the potential for sludge accumulation at the flow distribution baffle.

3.1.1.2 Primary Side Stress Corrosion Cracking

D.C. Cook Unit 2 steam generator tubes suffered primary side SCC of tight radius U-bend tubes at the tangent points and at the apex of row 1 tubes. This phenomenon has occurred in many steam generators because mill annealed Inconel 600 is susceptible to corrosion and the high residual stresses resulting from the bending operation for the short bend radii of the row 1 and row 2 tubes can result in primary site stress SCC. The use of corrosion resistant I-690 TT in lieu of I-600 MA is intended to improve the tubing's resistance to primary side SCC. The replacement lower assemblies will have a minimum U-bend radius of 3.14 inches compared to 2.19 inches for the original assemblies, thereby reducing the residual stresses from bending. In addition, the U-bend regions of the eight innermost tube rows of the replacement tubing will be stress relieved after bending to further reduce residual stresses.

Primary side stress corrosion cracking in the tube sheet area is expected to be reduced since expansion of the tubes into the tube sheet will be performed hydraulically. Hydraulic expansion reduces the residual tube stresses in the transition zone between the expanded and unexpanded sections of the tube and this reduction of residual stresses should result in lower primary side stress corrosion cracking in the tube sheet area.

3.1.1.3 Wear at Antivibration Bar Locations

Fretting wear resulting from excessive tube vibration has occurred in the U-bend region at tube-to-antivibration bar intersections. This wear has been attributed to several causes. Chrome-plated Inconel bars do not provide the best wear in combination with the Inconel tubes. The free span length of some tubing is quite large in larger radius U-bends. The natural frequency of the tubing is lower for the longer free span lengths. In addition, the design and manufacturing tolerances employed for the tubing and the AVBs sometimes result in excessive gaps at some locations, which increases the potential for wear.

The utility claims that the replacement steam generator AVB design and installation will minimize the wear potential. Three sets of AVBs will be used in the replacement steam generator tube bundle U-bend region. The original design had only two sets of AVBs. The AVBs will be wider and fabricated from SA-240 Type 405 stainless steel, which should generate less tubing wear than the chrome plated AVBs used in the original assemblies. The manufacturing tolerances in the replacement steam generator design will be controlled to reduce tube-to-AVB clearances and thereby reduce wear.

3.1.2 Design Improvements to Increase Performance

Several modifications and refinements have been incorporated into the replacement steam generator design to improve the generator thermal hydraulic characteristics. The utility expects these improvements to enhance the steam quality and reduce the pressure drop in the primary loop.

3.1.2.1 Upper Assembly Internals Modifications

In order to increase the quality of the steam exiting the steam nozzle, additional secondary separator drains and middle deck relief openings will be added to improve the drainage of separated water in the steam drums. Steam vents will also be installed inside the existing access openings in the middle deck plates to minimize the re-entrainment of separated water.

3.1.2.2 Tube End Conditions

The replacement lower assemblies' tubes will be flush with the tube hole opening when welded to the tubesheet cladding. The elimination of protruding tube ends and tube fillet welds reduces entry pressure losses and results in a lower pressure drop in the primary loop.

3.1.2.3 Controlled Tube Wall Thickness

The replacement steam generator tubes will have a maximum average wall thickness of 0.05 inches. In order to achieve optimal thermal hydraulic performance the tubing will be procured with stringent wall thickness tolerance controls.

3.1.3 Design Improvements to Enhance Maintainability and Reliability

Based on past operating experience several design changes have evolved to enhance maintainability and reliability of the replaced steam generators. These improvements are expected to reduce erosion and corrosion and facilitate access to and inspection of the generators.

3.1.3.1 Feedring with J-Nozzles

New feedrings with Inconel J-nozzles will be installed. Inconel J-nozzles are expected to reduce the potential for erosion of the nozzles and hence improve the reliability of the steam generators.

3.1.3.2 Tube End Welds and Tubesheet Marking

The tube ends will be welded flush with the tubesheet cladding, thereby minimizing locations for crud buildup. The tube end locations will be marked on a 2x2 row by column pattern to facilitate tube identification.

3.1.3.3 Access Ports

The replacement lower assemblies will have six 6-inch access ports, two more than the original steam generators. The access ports are to be located above the tubesheet, which will enhance sludge lancing and inspection of the tubesheet areas.

3.1.3.4 Inspection Ports

There will be two 4-inch inspection ports in the transition cone lower shell at an elevation slightly above the top tube support plate. Therefore, the row 1 tubes will be directly observable. These inspection ports will facilitate inspection of the top tube support plate and the U-bend area of the tubing.

3.2 Materials of Construction

The repaired steam generator will have mechanical and thermal characteristics consistent with the original design and safety analysis as presented in the FSAR.

All materials used in the fabrication of the replacement lower assemblies will be identical to the original lower assemblies except for the following material changes.

- o Transition cone and stub barrel material has been changed from ASME SA-533 Grade A Class 1 to ASME SA-508 Class 3.

- o Tubesheet forging material has been changed from ASME SA-508 Class 2 to ASME SA-508 Class 2A.

- o Support plate material has been changed from ASME SA-285 Class C to ASME SA-240 Type 405.

- o Steam generator tube material has been changed from ASME SB-163 Alloy 600 to ASME SB-163 Alloy 690 (Code Case N-20).

3.3 Codes and Standards

The original steam generators were designed, fabricated, inspected and tested as Class A components in conformity with the 1968 ASME Boiler & Pressure Vessel Code, Section III plus Addenda through Winter 1968. All pressure boundary materials and weld filler materials conformed to specifications set forth by Section III of the ASME Code. Non-pressure retaining parts on the secondary side were in accordance with applicable ASTM or ASME material specifications.

The design, material, fabrication, inspection, examination, and testing of the replacement steam generator lower assemblies and components supplied by Westinghouse will be in accordance with the codes and standards, including all applicable addenda, as listed below:

3.3.1 Industry Codes and Standards

- o 1983 ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," plus Addenda through Summer 1984.

- o 1983 ASME Boiler and Pressure Vessel Code, Section IX, "Qualification Standards for Welding and Brazing Procedures, Welders, Brazers, and Welding and Brazing Operations," plus Addenda through Summer 1984.

- o 1983 ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications," plus Addenda through Summer 1984.

- o 1983 ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspections of Nuclear Power Plant Components," plus Addenda through Summer 1983 and applicable Code Cases and Interpretations.

- Appendix A of 10 CFR 50, as amended and effective October 27, 1978.

- Appendix B of 10 CFR 50, as amended and effective January 20, 1975.

- Appendix G of 10 CFR 50, as amended and effective May 27, 1983.

- State of Michigan Boiler Law and Rules and Regulations, as administered by the Michigan Department of Labor, Bureau of Safety and Regulation.

- ASTM A262-84 "Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels."

In addition the original "N" stamp will be maintained for the repair steam generators. The replacement lower assemblies will be "NPT" stamped and the field closure welds will be "NA" stamped.

3.3.2 NRC Regulatory Guides and Codes

The following NRC Regulatory Guides will also be applicable to the replacement steam generator project:

- Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" (Rev. 3, April 1978).

- Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (March 1973).

- Regulatory Guide 1.44, "Control of Sensitized Stainless Steel" (Rev. 0, May 1973).

- Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel" (Rev. 0, May 1973).

- o Regulatory Guide 1.85, "Material Code Case Acceptability - ASME Section III Division I"" (Rev. 24, July 1986). Code Case N-20 of Regulatory Guide 1.85 which covers the use of Inconel 690 is the only Code Case that will be applied.

3.3.3 Fabrication Alternatives

Westinghouse, the fabricator of the replacement lower assembly, states that it cannot comply with Regulatory Position C.2 of R.G. 1.50, which recommends that "preheat temperature should be maintained until a post-weld heat treatment has been performed." Westinghouse states that it is not possible, due to the size and weight of component assemblies and subassemblies and the configuration of the post-weld heating furnace, to maintain preheat until post-weld treatment begins. In lieu of this practice, Westinghouse procedures require a "hydrogen bake" cycle (raising preheat to minimum 400^{o}F and holding for minimum 4 hours) prior to lowering temperature to ambient.

3.4 Shop Tests and Inspections

The tests and inspections required by the ASME Code, Section III will be conducted during shop fabrication of the replacement steam generator lower assemblies. In addition to the ASME requirements, further tests and inspections will be conducted at the fabrication facility. These will involve a gas leak test to demonstrate the integrity of the tube-to-tubesheet welds and a primary side hydrotest in accordance with approved procedures.

3.5 On Site Installation

Reinstallation (welding) of piping and of the steam generator shell weld will be performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III (1983 Edition plus Addenda through Summer 1984) and Section XI (1983 Edition plus Addenda through Summer 1983). The reinstalled piping system will meet FSAR criteria.

Weld qualification will be in accordance with Section IX plus requirements of Section III. Weld filler metals and electrodes will be ordered in accordance with Section II Part C, and covered electrodes will meet the analysis tests of Section III, NB-2420. Austenitic stainless steel welds will meet the delta ferrite requirements in Section III, NB-2433. Nondestructive examination (NDE) of welds will be conducted in accordance with Section V of the Code and NDE acceptance standards will meet Section III requirements.

There will not be any field welded joints between clad components. The replacement steam generator channel head cladding will be applied at the fabrication facility. Safe ends will be installed on the inlet and outlet reactor coolant nozzles at the fabrication facility so that field welding of the nozzles to the stainless steel pipe fittings can be performed without the need for stress relief heat treatment.

3.6 Post-Installation Inspections and Tests

The primary side of each steam generator head will be visually inspected for dirt and debris for removal prior to return to service. The secondary side will also be inspected and thoroughly cleaned to remove debris or foreign objects.

Baseline inspection records will be established to meet ASME Code Section XI requirements. These records will include 100% eddy current tube examinations. Field radiography of the main steam and feedwater lines and ultrasonic testing of the primary reactor coolant pipe and steam generator welds will also be provided as baseline inservice inspection records.

3.7 Electropolishing and Passivation

The NRC staff asked Indiana and Michigan Electric Power Co. to determine the feasibility of using electropolishing/passivation to condition the primary surfaces of the steam generator channel head. The objective of this treatment would be to inhibit the deposition of radioactive crud from the primary system onto the channel head surfaces and thereby lower the occupational exposure to workers performing eddy current testing and tube repairs.

Indiana and Michigan Electric Power Co. reviewed industry literature and consulted with the steam generator manufacturer (W) to determine the state-of-the-art and industrial experience with respect to performing electropolishing/passivation of the steam generators. Based on their findings, Indiana and Michigan Electric Power Co. could not recommend performing electropolishing/passivation. The steam generator manufacturer indicated that it is not capable of performing these processes in the shop prior to steam generator shipment. While field processing is being evaluated, Indiana and Michigan Electric Power Co. assessments indicate that these processes are not fully qualified (for this application), not economically justified and not endorsed by the steam generator manufacturer.

4. EVALUATION

The proposed repair/replacement program for the D.C. Cook Unit 2 steam generators addresses those aspects of materials selection, design and secondary system operation where operating experience indicates that deficiencies existed. These deficiencies resulted in tube degradation and poor performance of the original steam generator tubes leading to excessive repairs and eventual replacement. The licensee proposes materials and design changes to minimize tube degradation and to improve performance, reliability, and maintainability.

The use of Inconel 690 thermally treated tubing in lieu of the original Inconel 600 mill annealed tubing is the most significant material change in the replacement steam generator lower assemblies and the staff agrees that this would improve corrosion resistance of the repaired steam generators. Corrosion tests under Electric Power Research Institute (EPRI) sponsorship have indicated that Inconel 690 TT has a negligible corrosion rate. Accelerated stress corrosion tests in caustic and chloride aqueous solutions have also indicated that Inconel 690 TT resists general corrosion in the aggressive environments. Isothermal corrosion tests in high purity water have shown that, at normal stress levels, thermally treated Inconel 690 with normal microstructure has high resistance to intergranular stress corrosion cracking in extended high temperature exposure. EPRI concludes, as a result of these laboratory corrosion testing, that Inconel 690 TT material should be used for PWR steam generator tubing with AVT secondary water systems (Ref. 1). Inconel 690 is a Code approved material (ASME SB-163) and covered by ASME Code Case N-20 and accepted by the NRC under Regulatory Guide 1.85 (Rev. 24, July 1986).

The substitution of SA-240 Type 405 stainless steel for carbon steel support plates should minimize tube degradation resulting from denting in the tube-tube support plate region. Corrosion oxides from the carbon steel support plate build up in the crevice between the support plate and the tubes and lead to denting of the steam generator tubing in that area. Corrosion of SA-240 results in an oxide that has approximately the same volume as the parent material, whereas corrosion of carbon steel results in oxides (magnetite) that have a larger volume than the parent material. The selection of SA-240 Type 405 ferritic stainless steel as a replacement for carbon steel support plates is based on EPRI sponsored corrosion studies (Ref. 2). In addition, SA-240 has a low wear coefficient when paired with Inconel and has a coefficient of thermal expansion similar to the carbon steel it will replace.

The licensee's contention that the replacement tube support plate quatrefoil hole design (four flow holes and four support lands) will provide improvement in steam generator tube performance by preventing sludge deposition and the potential for corrosion at support plate locations is acceptable in that a high sweeping flow through the four flow holes prevent sludge build-up.

The licensee's contention that stress corrosion cracking at transition regions of the tube in the tubesheet would be minimized by changes in replacement assembly manufacturing design is also acceptable, based on the following argument. The new lower assemblies will have the tubes hydraulically expanded for the full depth of the tubesheet. Full depth hydraulic expansion of the tubes in the tubesheet holes will eliminate the tubesheet crevice in which a high concentration of impurities can accumulate. An additional benefit of the hydraulic expansion process is the reduction of cold working from that caused by mechanical hard rolling (as used in the original design). This results in lower residual stresses at the transition of the expanded to the unexpanded region of the tubes. Because tensile stresses in hydraulically expanded tubes may be half of the stresses encountered in mechanically rolled tubes, the potential for stress corrosion cracking should be reduced.

The licensee's contention that stress corrosion cracking at the short radius U-bend regions would be reduced by the replacement assembly is also acceptable to the NRC staff, based on the following. The U-bend regions of the eight innermost tube rows will be stress relieved after bending. Additionally, bending stresses caused by manufacturing will be reduced in row 1 tubes by increasing the row 1 U-bend radius.

The use of three instead of two sets of AVBs should provide additional support to the tubes in the U-bend region thereby reducing the potential for tube vibrational instabilities that lead to tube wear. The wear potential between the AVBs and the Inconel 690 tubes should also be reduced by the use of SA-240 Type 405 stainless steel AVBs in place of the original chrome plated AVBs because there is better wear compatibility between Type 405 stainless steel and Inconel 690 than between hard chrome plating and Inconel 690.

The NRC staff has previously accepted Westinghouse's alternative to Position C.2 of Regulatory Guide 1.150 which states that preheat temperature should be maintained until post-weld heat treatment has been performed. As indicated in Section 3.3.3, Westinghouse's alternative to Position C.2 is to perform a "hydrogen bake" cycle (minimum 400^{o}F for 4 hours) prior to lowering the temperature to ambient, followed by post-weld heat treatment. Westinghouse has described its alternative position in Topical Report - WCAP-8577, "The Application of Preheat Temperatures after Welding Pressure Vessel Steels," September 1975. Based on tests and technical data presented by Westinghouse, the NRC staff concluded in their review of the Topical Report dated June 18, 1976 that Westinghouse procedures and controls provide reasonable assurances that cracking of components made from low alloy steels will not occur during fabrication and the possibility of subsequent cracking due to residual stresses in the weldment is minimal.

The licensee's contention that electropolishing/passivation of the channel head primary surfaces to minimize radioactive corrosion product buildup is not a viable option at present is acceptable, based on the following argument. Electropolishing is the process of smoothing a metal surface anodically in a concentrated acid or alkaline solution (Ref. 3). A typical electropolishing electrolyte for smoothing the channel head surface would be a sulfuric-phosphoric-chromic acid electrolyte which must be rigorously cleaned after use to prevent subsequent damage to welds or entrapment in crevices. Since electropolishing of the steam generator channel head surfaces has not been used in nuclear systems it does not appear to be warranted for use in the D.C. Cook Unit 2 repair program.

5. CONCLUSIONS

The Indiana and Michigan Electric Power Co.'s report entitled "Steam Generator Repair Report" for Donald C. Cook Nuclear Plant, Unit No. 2 describes the safety related aspects associated with the repair of steam generators by replacement of the lower assembly (tube bundle) of the existing units with shop fabricated replacement lower assemblies. Design improvements, material substitutions, applicable codes and standards, and proposed testing, installation and inspection plans have been reviewed and evaluated.

Based on this evaluation it has been concluded: (1) that changes in steam generator mechanical design, thermal hydraulics, materials selection, tube fabrication techniques, and changes in secondary systems design and operation appear to be effective solutions to steam generator problems previously encountered at D.C. Cook Unit 2 and (2) that the unit may be repaired and operated without undue risk to public health and safety.

REFERENCES

1. EPRI - Steam Generator Owners' Group, "Steam Generator Reference Book," May 1, 1985.

2. EPRI NP-3924-SR, "The Second EPRI Workshop on Support Structure Corrosion in Nuclear Plant Steam Generators," March 1985.

3. Metals Handbook, Vol. 2, "Heat Treating, Cleaning and Finishing," 8th Edition, 1964.