March 8, 1988

Docket No. 50-316

DISTRIBUTION

Docket File	TBarnhart(4)
PD31 Plant Gray	Wanda Jones
NRC & Local PDRs	OGC
GHolahan	EButcher
RIngram	ACRS (10)
JStang	GPA/PA
EJordan	ARM/LFMB
DHagan	JPartlow

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

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. DPR-74: STEAM GENERATOR REPAIR PROGRAM (TACS NOS. 63997, 65113 AND 65114)

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of a change to the license in response to your application dated March 12, 1987.

The amendment approves the steam generator repair program for Donald C. Cook, Unit 2, and provides a license condition related to the repair program.

Copies of the Safety Evaluation and the notice of issuance are also enclosed.

Sincerely,

Original signed by

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 100 to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRSP	PM/PD31:DRSP	D/PD31:DRSP	OGC
RIngram	JStang:jlt	MVirgilio	
3/2/88	3/3/88	3/8/88	/ /88

OFFICIAL RECORD COPY

8803140360 880308
PDR ADOCK 05000316
P PDR

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgeman, Michigan 49106

W. G. Smith, Jr., Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mayor, City of Bridgeman
Post Office Box 366
Bridgeman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

Mr. J. Feinstein
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated March 12, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Paragraph 2.I is added to Facility Operating License No. DPR-74 to read as follows:

2.I <u>STEAM GENERATOR REPAIR PROGRAM</u>

(1) The licensee is authorized to repair Unit 2 steam generators by replacement of major components. Repairs shall be conducted in accordance with the licensee's commitments identified in the Commission approved Donald C. Cook Nuclear Plant Unit No. 2 Steam Generator Repair Report dated November 7, 1986, as revised through Revision 6, and additional commitments identified in the staff's related Safety Evaluation dated .

(2) The Technical Specifications identified in Table 3.2-2 of the Steam Generator Repair Report dated November 7, 1986, as revised through Revision 6 dated February 18, 1988, will not be applicable during the repair program. For purposes of Technical Specification applicability, the Steam Generator Repair Project will begin when the last fuel assembly from the Unit 2 core is placed in the spent fuel pool and will end when the first fuel assembly is removed from the spent fuel pool to refuel the Unit 2 core.

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

FOR THE NUCLEAR REGULATORY COMMISSION

[signature] for

Martin J. Virgilio, Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Date of Issuance: 3/8/88

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.100 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated March 12, 1987, the Indiana Michigan Power Company (the licensee) requested an amendment to Facility Operating License No. DPR-74. The amendment would permit the repair by replacement of major components in all four steam generators of Donald C. Cook Nuclear Plant, Unit No. 2.

The granting of the proposed amendment would allow the licensee to repair the four steam generators in Unit No. 2 in accordance with the Steam Generator Repair Report (the Report) submitted by the licensee by letter dated November 7, 1986, as revised by the licensee on March 30, July 24, October 20, and December 4, 1987, and February 18, 1988. The four steam generators have incurred intergranular attack/stress corrosion cracking of tubes in the tube-sheet region and in the tube support regions. The tubesheet region cracking could be corrected somewhat by sleeving the tube ends; however, the tube support regions can be corrected only by plugging all the affected tubes. As of November 1986, the licensee has plugged 763 tubes and, to limit further chemical corrosion, has initiated an on-line boric acid treatment of the steam side of the tubes and is administratively limiting the unit to 80% power; this latter effort is to further reduce the high temperature sensitive, corrosion attack. As a result of continued tube degradation, the associated penalty of reduced generating capacity, and the lack of practical repair techniques, the licensee is proposing to replace the half of each of the steam generators containing the tubes. The licensee will take advantage of the required replacement to also refurbish the remaining half of each steam generator to ultimately improve and enhance steam generator and plant performance.

The repair program will involve partial disassembly of the reinforced concrete enclosures surrounding the steam generators, cutting and removing portions of the steam lines and feedwater lines, cutting the inlet and outlet reactor coolant lines, cutting the steam generators in half to preserve and refurbish the upper assemblies and to replace the lower assemblies (all tubes, tube supports, and tubesheet), moving and storing the used lower assemblies on-site until eventual plant decommissioning, and reassembling the unit with new lower assemblies in the reverse order. The repaired steam generators will be similar in design and functionally the same as the original steam generators.

Westinghouse will fabricate the replacement steam generator lower assemblies. The repaired steam generators will be similar in design and will be functionally the same as the original steam generators. The design 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.

To accomplish this effort, the licensee will improve several existing roads to facilitate movement of contractor personnel on-site and to accommodate the movement of the lower assemblies to the storage area, build a new storage building to house the used lower assemblies until decommissioning, build a new site security access building and a containment access building to facilitate contractor access as required, and erect a fabrication shop/warehouse for repair, storage, and training. All of these activities will be accomplished within areas already impacted by plant construction and operation.

2.0 BACKGROUND

Donald C. Cook Unit 2 incorporates a nuclear steam supply system manufactured by Westinghouse and is licensed for 3411 MWt. Initial criticality occurred on March 10, 1978. The unit completed its fifth fuel cycle on February 28, 1986, at which time about 5.3 effective full power years of operation had been accrued.

Donald C. Cook Unit 2 has four Westinghouse Series 51 steam generators, each having 3388 tubes, 0.875 inch O.D. by 0.005 inch thick; tubing material is Inconel 600 in the mill annealed condition. The tubes are hardrolled for a distance of approximately 2.25 inches above the bottom of the tubesheet, leaving an open annular crevice of approximately 18.75-inch depth and 7- to 9-mil radial gap. Tube support plates are carbon steel with drilled tube holes having radial clearance of approximately 8-mils.

The first significant indication of secondary side corrosion of Donald C. Cook Unit 2 came in late 1983. During startup on November 7, 1983, following an outage to plug leaking Row 1 tubes, there were immediate indications of primary-to-secondary leakage in steam generator 21, and the unit was removed from service. Visual inspection of the primary side of the tubesheet under a static head of water showed the hot leg of tube R16C40 to be leaking. Subsequent eddy current testing (ECT) of about 725 tubes in steam generator 21 revealed the defect in R16C40 to be just above the secondary face of the tubesheet, and indicated two additional tubes, R14C40 and R14C41, with similar tube degradation. In addition, ECT of over 500 tubes in steam generator 22 was performed; no degradation was found. The unit was restarted on November 22, 1983, and ran until March 10, 1984, when it was removed from service for refueling.

Steam generator activities during that refueling outage included complete ECT of all four steam generators and removal of sections of seven tubes for metallurgical analysis. ECT resulted in plugging 61 tubes due to indications of tube degradation at or just above the tubesheet surface and five tubes in the tubesheet crevice region. Metallography confirmed that degradation was due to intergranular attack/stress corrosion cracking (IGA/SCC) caused by caustic environment. Unit 2 was restarted on July 7, 1984, and ran without steam generator related problems until it was removed from service on July 15, 1985, with an indicated leak of 0.22 gpm in steam generator 23.

Visual inspection under a static head of water showed one leaking tube (R16C56) in steam generator 23. Helium leak detection revealed no other leakage. ECT of the leaking tube indicated a defect approximately one inch below the top of the tubesheet. Additional ECT of a block of 24 tubes around tube R16C56 revealed tube R15C55 to have a similar defect. Reanalysis of ECT data taken in 1984 showed that tube R15C55 had a 20 percent through-wall indication that was not identified at the time. Tubes R15C55 and R16C56 were plugged.

Donald C. Cook Unit 2 was restarted on August 2, 1985. During startup, radiation monitors on the condenser air ejectors and samples of steam generator blowdown indicated slight additional leakage in steam generator 23. The unit was again removed from service. ECT of approximately 1500 tubes in the sludge pile region of the tubesheet was conducted. As a result, 35 tubes were plugged, many of which had no previous indication of degradation. Due to concern over the apparent pervasiveness of IGA/SCC in the tubesheet region, a boric acid soak was performed at 30 percent power with 50 to 55 ppm boron concentration, followed by on-line boric acid addition to maintain a 5 to 10 ppm boron concentration. Addition of boric acid was intended to neutralize the alkaline environment and thus possibly slow the rate of IGA/SCC.

Donald C. Cook Unit 2 was again restarted, but on August 23, 1985, during a hold at 30 percent power for a boric acid soak, a 0.20 gpm steam generator leak was detected. All four steam generators were opened and visually inspected under a static head of water. Two leaking tubes were identified in steam generators 22 (R14C41) and 24 (R19C52). ECT was expanded to include all the tubes in all steam generators. This inspection resulted in plugging an additional 110 tubes. For the first time, evidence of tube corrosion at hot leg support plate intersections was indicated by ECT. Because the condition of the tubes at support plate intersections could influence any future decision to repair the tubesheet region by sleeving, five tube samples were removed to assess the condition of the tubes at support plates. Analysis of the tube samples confirmed the presence of axially oriented intergranular stress corrosion cracks at support plate locations.

Following the complete ECT, Donald C. Cook Unit 2 was successfully returned to service in October 1985, and ran without significant steam generator related problems until removed from service for refueling on February 28, 1986. At the time of shutdown, steam generator primary-to-secondary leakage was in the range of 0.001 to 0.04 gpm; hydrostatic testing at 600 psig revealed one leaking tube (R16C45) in steam generator 22. Subsequent ECT showed the defect to be located about one inch below the hot leg tubesheet. ECT of all affected areas of all four steam generators resulted in plugging 151 tubes.

To date the total number of tubes plugged in each steam generator is as follows: steam generator 21, 142 tubes; steam generator 22, 210 tubes; steam generator 23, 210 tubes; and steam generator 24, 201 tubes.

The steam generators at the Donald C. Cook Unit 2 have experienced corrosion-related phenomena, as discussed above, which have required periodic examination and plugging of steam generator tubes to ensure continued plant operation. At present, Donald C. Cook Unit 2 is continuing to receive on-line boric acid treatment and is being administratively limited to 80% power to retard the rate of steam generator tube degradation. However, the need to plug additional tubes may continue, and the Technical Specification plugging limit may be reached in the future. This could lead to the need for additional Technical Specification changes and possibly result in permanent power limitations.

As a result of continued tube degradation, the associated penalty of reduced generating capacity, and the lack of practical repair techniques, the licensee is proposing to replace the Donald C. Cook Unit 2 steam generator lower assemblies. This replacement would repair the steam generators and allow the unit to operate at its design capacity.

3.0 SCOPE OF WORK TO BE PERFORMED

The steam generator repair project proposed by the Indiana Michigan Power Company for the Donald C. Cook Unit 2 is similar to the steam generator repairs completed by the Virginia Electric and Power Company for the Surry Power Station and by Wisconsin Electric Power Company for the Point Beach Nuclear Plant Unit 1. Both projects utilized the reactor coolant system pipe-cutting process to remove the steam generator lower assemblies as proposed by the licensee for the Donald C. Cook Unit 2.

Repair of the Donald C. Cook Unit 2 steam generators will involve the complete replacement of the steam generator lower assemblies. An opening will be cut in the reinforced concrete doghouses surrounding the steam generators to provide access. To facilitate removal, each steam generator will be cut on the upper assembly shell plate just above the transition cone girth weld and at the inlet and outlet reactor coolant piping nozzles. The steam line piping and feedwater piping will be cut and sections removed. The steam generator upper assembly will be lifted off and removed from containment to undergo internal modifications to the moisture separation and feedring equipment. The steam generator lower assemblies will then be lifted from their supports and transported out of containment to the temporary on-site steam generator storage facility. The removal route for both the upper and the lower assemblies will be through the equipment hatch at the 650' elevation and through the auxiliary building to the railroad bay. The containment equipment hatch is sized to accommodate steam generator replacement without containment modification. The existing polar crane will be utilized for all heavy lifting inside of the containment building. The replacement steam generator lower assemblies will be transported and installed in a similar manner. The original upper and new lower assemblies as well as all associated piping will be welded together in the field.

All components and piping will be reinstalled to meet the original design configurations and installation requirements, thus eliminating any design modifications which would require changes to the original design analysis.

4.0 REPLACEMENT STEAM GENERATOR DESIGN

The repaired 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.

4.1 Materials of Construction

The repaired steam generators will have mechanical and thermal characteristics consistent with the original design and safety analysis as presented in the Final Safety Analysis Report (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.

- Transition cone and stub barrel material has been changed from ASME SA-533 Grade A Class 1 to ASME SA-508 Class 3.
- Tubesheet forging material has been changed from ASME SA-508 Class 2 to ASME SA-508 Class 21.
- Support plate material has been changed from ASME SA-285 Class C to ASME SA-240 Type 405.
- Steam generator tube material has been changed from ASME SB-163 Alloy 600 to ASME SB-163 Alloy 690 (Code Case N-20).

4.2 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:

4.2.1 Industry Codes and Standards

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

- 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 repaired steam generators. The replacement lower assemblies will be "NPT" stamped and the field closure welds will be "NA" stamped.

4.2.2 NRC Regulations and Guides

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).

- 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.

- Appendix B of 10 CFR 50.

4.2.3 Fabrication Alternatives

Westinghouse, the fabricator of the replacement lower assembly, states that it cannot comply with Regulatory Position C.2 of Regulatory Guide 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 temperature until post-weld treatment begins. In lieu of this practice, Westinghouse procedures require a "hydrogen bake" cycle (raising preheat to a minimum of 400°F and holding for a minimum of 4 hours) prior to lowering temperature to ambient.

The NRC staff has previously accepted Westinghouse's alternative to Position C.2 of Regulatory Guide 1.50 which states that preheat temperature should be maintained until post-weld heat treatment has been performed. As indicated above, Westinghouse's alternative to Position C.2 is to perform a "hydrogen bake" cycle (minimum 400°F for 4 hours) prior to lowering the temperature to ambient, followed by post-weld heat treatement. 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 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.

4.3 Evaluation/Conclusion

The proposed repair/replacement program for the Donald 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 licensee'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 system design and operation will be effective solutions to steam generator problems previously encountered at Donald C. Cook Unit 2, and (2) that the unit may be repaired and operated without undue risk to public health and safety.

5.0 QUALITY ASSURANCE

The Donald C. Cook Unit 2 steam generator repair project will have a quality assurance program that ensures compliance with the applicable regulatory requirements. The Donald C. Cook FSAR Chapter 1.7, "Quality Assurance" also referred to as the "Updated Quality Assurance Program Description for the D. C. Cook Plant" or "QAPD," supplemented with a Steam Generator Repair Quality Assurance Program (SGR QAP) will administer the requirements. Topics in this report that will be expanded in the SGR QAP are:

- Organization
- Quality Assurance
- Design Control
- Procurement Document Control
- Audit and Surveillance
- Document and Record Control
- Nonconformance and Corrective Action Control

Section 5 of the "Steam Generator Repair Report," (SGRR) through revision 5, issued December 4, 1987, describes the quality assurance program applicable to the safety-related activities of the steam generator repair project. In addition, it states that the Donald C. Cook plant QAPD, supplemented with a SGR QAP, will administer the requirements.

The program is applicable to all safety-related activities to be conducted for the steam generator repair including disassembly, removal, site fabrication, installation, inspection, and return-to-service testing. Manufacture of new steam generator lower assemblies by the NSSS, as well as fabricated material and equipment of other suppliers, are all accomplished with their own QA programs approved by AEPSC Corporate QA in accordance with this controlling program.

Applicable industry codes and standards along with related NRC regulatory guides are included in Section 3 of the SGRR and in Appendices A and B of the QAPD. Additional exceptions/clarifications regarding structural concrete testing are located in Appendix B of the SGR QAP (Items 7c and 7d).

Overview auditing, surveillance, and assessment activities are provided for within the program and results are being provided to appropriate management.

The staff has reviewed the QA program as described in Section 5 of the SGRR and in addition reviewed the final draft version of the SGR QAP. The program and its application was discussed with licensee personnel at the Donald C. Cook plant site on December 1-3, 1987. Acceptance in part is based on the fact that previously accepted program commitments will be applied to the steam generator repair or that revised commitments are no less stringent than the previously existing commitments.

Based on the staff reviews of the entire program, site discussions and evaluations, the staff concludes that the quality assurance program designated as applicable to the steam generator repair at Donald C. Cook Unit No. 2 is acceptable.

6.0 PREVENTION OF LOOSE PARTS

Loose parts and foreign objects left inside the steam generators have been identified as the cause of at least two steam generator tube rupture events. Past inspections have found a variety of foreign objects in the secondary side of the steam generators. Approved procedures and/or specifications will be followed by the licensee during all cutting operations to prevent debris or cutting chips from entering piping systems or the reusable pipe sections and to maintain overall cleanliness. Where possible, dams will be employed in piping systems to minimize ingress of cutting chips or slag. Where dams cannot be used, cutting methods, which will minimize chips on the final parting cuts, will be considered. After all cutting operations, the system piping and removed sections will be cleaned and capped.

Approved procedures and/or specifications will incorporate the requirements of N45.2.1-1973 and Regulatory Guide 1.37, March 1973.

Work areas within the reactor coolant system will be sealed prior to commencement of repair activities and then thoroughly cleaned before being returned to service to minimize the need for flushing.

The primary side of each steam generator channel head will be inspected. Any dirt and debris will be removed prior to return to service. The secondary side of the steam generators will be inspected before being returned to service and thoroughly cleaned to remove debris or foreign objects.

7.0 RADIOLOGICAL CONSIDERATION

7.1 Background

The NRC staff has evaluated the radiation protection measures established by the licensee for the steam generator repair program at Donald C. Cook Nuclear Plant, Unit No. 2, including those features intended to ensure that doses will be maintained as low as is reasonably achievable (ALARA). The licensee has estimated that the total occupational exposure from the proposed four steam generator (SG) repair program will be 1733 person-rem. On a per-steam generator basis, this compares favorably with previous repair programs. The licensee's person-rem estimate per steam generator for Cook Unit 2 is 433 as compared to actual exposure at Surry 1 (586), Surry 2 (714), Turkey Point 3 (717), Turkey Point 4 (435), and H. B. Robinson (420). On the basis of the staff's review of: (1) the licensee's reports, dated November 4, 1986, March 30, 1987, July 24, 1987 and October 20, 1987, (2) licensee's response to the staff's request for additional information (September 25, 1987), (3) health physics inspections conducted at Donald C. Cook by NRC Region III (October 23, 1987), (4) operational experiences from other repair programs, and (5) ALARA guidelines that reflect past steam generator repair programs, the staff concludes that the licensee's estimate of 1733 person-rem to the work force is a reasonable estimate of the expected dose.

7.2 Occupational Exposure

The repair program will involve partial disassembly of the reinforced concrete enclosures surrounding the steam generators, cutting and removing portions of the steam lines and feedwater lines, cutting the inlet and outlet reactor coolant lines, cutting the steam generators in half to preserve and refurbish the upper assemblies and to replace the lower assemblies (all tubes, tube supports, and tubesheet), moving and storing the used lower assemblies on-site until eventual plant decommissioning, and reassembling the unit with new lower assemblies in the reverse order. The dose estimated by the licensee for the generator replacement task at Donald C. Cook Unit 2 is 1733 person-rem. The licensee's estimates were derived from anticipated person-hours in known radiation fields for all tasks planned. This methodology is acceptable, and the dose estimates are consistent with doses observed for similar steam generator repair work in the industry. Dose expended is expected to be offset by dose reductions and savings over the next 10 years through successful completion of the project.

7.3 ALARA Consideration

The licensee's total estimate of 1733 person-rem for the steam generator repair program takes into account the dose reduction measures described in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." The licensee will utilize methods that minimize individual and collective doses through the use of preplanning, access control, personnel monitoring, radiation and contamination surveillance, airborne radioactivity control, radwaste control, training, pretask dose assessment, source reduction via decontamination and shielding, and evaluation of similar steam generator replacement experiences (Surry, H. B. Robinson, Point Beach, and Turkey Point).

Some examples of the licensee's ALARA/radiation protection measures for the steam generator repair program are listed below:

(1) Shielding - Shielding in the containment will be based on special requirements (i.e., machine and equipment interferences), man-rem and cost benefit analysis, and Project Radiological Protection/ALARA Group requirements.

(2) Worker Surveillance - Radiation Protection Technicians will provide continuous work surveillance in the containment and other work areas by-on-the-job technician coverage and/or remote monitoring by closed circuit audio/video systems.

(3) Core Configuration - The core will be unloaded and the fuel stored in the spent fuel pool to help maintain doses ALARA.

(4) Steam Generator Secondary Side Water Level - Water level in the secondary side of the steam generator will be kept as high as possible to provide additional shielding.

(5) Dosimetry - Whole body badging, electronic dosimetry (remote and self-reading), extremity badging and/or standard self-reading pocket dosimetry will be used to monitor doses during the project. TLDs will provide the official dose record.

(6) Exposure Tracking and Trending - A computerized system will be used to track individual tasks, personnel, groups, and activities.

The licensee has reviewed outage sequences utilizing ALARA coordinators to determine the specific applications of the above measures. Laydown and "wait areas" have been or will be clearly identified and prepared for the task start, including provisions for decontamination, temporary shielding, posting, and access.

Radiation, containment, and airborne radioactivity surveys will be conducted as necessary to determine radiation protection measures. The licensee has an adequate number of portable air sampling instruments available for the replacement task as discussed in Regulatory Guide 8.8, Section C.4, and has verified that Donald C. Cook Unit 2 counting facilities will be adequate for the anticipated increased surveillance activities as in Section C.4.

The licensee will use respiratory equipment and the containment ventilation system to control airborne radioactivity. Engineering controls which preclude the need for respiratory protection equipment (e.g., contamination control devices, local HEPA (high-efficiency particulate air) ventilation, flexible ducting, and tents) as recommended in Regulatory Guide 8.8, Section C.2.d, will be utilized and specific applications will be identified before the start of the outage. A program for ALARA internal and external contamination as part of existing procedures will be provided consistent with Section C.2.d in order to reduce potential doses to workers who receive detectable internal contamination, as well as to minimize the number of workers who become externally contaminated.

Decontamination facilities adequate for the replacement task will be provided.

The licensee will also provide extensive training for workers that will emphasize ALARA measures. The licensee has committed that adequate training facilities and training personnel will be available to conduct the committed training before initiation of the related task for all persons working in radiation control areas. The training program to be conducted by the licensee includes measures to familiarize workers with their tasks, tools, equipment, and operational and radiological procedures by use of job-specific training, dry-run training, and mockup training.

Methods for handling and processing radioactive wastes, and the impacts of these wastes, have been evaluated. Radwaste reduction techniques and training have been planned for the outage. Additional information on the radioactive waste is provided in the Donald C. Cook Unit 2, "Environmental Assessment," November 23, 1987.

The licensee has committed to measure and evaluate the progress of the steam generator replacement task through dose tracking and ongoing radiological assessment of specific tasks by radiological engineers/ALARA coordinators as is recommended in Regulatory Guide 8.8, Sections C.1 and C.3.

The licensee has formed a Project Radiological/ALARA Group to provide ALARA engineering support and radiological controls for the steam generator repair project. This group is staffed by the corporate office and contractor personnel and will report to the Project Health Physicist, who will have overall responsibility for the ALARA and radiological protection coverage for the steam generator repair project. The Radiological Protection/ALARA Group is comprised of five support groups which report to the Project Health Physicist. The support group includes: (1) Radiation Project Group, (2) ALARA Group, (3) Dosimetry and Records Group, (4) Radiation Protection Support Services Group, and (5) the Training Group.

In order for the NRC staff to evaluate the radiological results of the replacement project, and to determine if additional or different radiological controls need to be considered, the licensee will perform a radiological assessment as follows:

(1) The collective occupational dose estimates will be updated each 90-day period. If the updated estimate exceeds the person-rem estimate by more than 10%, the licensee will provide a revised estimate, including the reasons for such changes, to the NRC with the 90-day Progress Reports.

(2) A final report shall be provided to the NRC within 60 days after completion of the repair. This report will include:

(a) a summary of the occupational dose received by major task, (b) a comparison of estimated doses with the doses actually received, (c) a discussion of ALARA measures employed, and (d) a summary of decontamination efforts and radwaste generation.

(3) Interim reports which summarize each 90-day period of the repair effort shall be provided to the NRC within 60 days of the completion of each such period.

7.4 Conclusion

Based on the above evaluation, the NRC staff concludes that the programs and procedures proposed by the licensee in making the steam generator repairs demonstrate that it will meet the requirements of (1) 10 CFR Part 20 limits and as it relates to effort to maintain radiation exposure as low as is reasonably achievable; (2) Regulatory Guide 8.8, as it relates to management policy and organization; personnel qualifications and training; design of facilities and equipment; radiation protection program, plans, and procedures; and the availability of supporting equipment, instrumentation, and facilities; (3) Regulatory Position C.1.f of Regulatory Guide 8.10 on modifications to reduce radiation exposures. The staff, therefore, finds the licensee's plan to be acceptable.

8.0 HANDLING OF HEAVY LOADS

The existing polar crane will be utilized for handling heavy loads inside containment during the steam generator repair project. The licensee indicates that no work regarding the steam generator repair will be undertaken until all fuel is removed from the reactor vessel and placed within the spent fuel pool, and the reactor coolant loops drained. Thus, potential offsite dose considerations resulting from the dropping of heavy loads during the repair phase are not applicable. Further, any consequences from a load drop would be of an economic nature and not a radiological safety concern.

The polar crane is equipped with a 250-ton capacity main hoist and 35-ton auxiliary hoist mounted on a single trolley. The polar crane possesses sufficient capacity to handle all major lifting requirements for the steam generator project inside containment and can be rerated to a higher capacity as required; however, rerating of the hoist is not anticipated.

Movement of the very heavy steam generator lower assemblies through the auxiliary building and over a portion of the spent fuel pool was very carefully assessed by the licensee and very carefully reviewed by the NRC staff.

To handle the steam generator lower assemblies in the auxiliary building, a new auxiliary building crane will be installed and the existing auxiliary building crane, which is in the process of being upgraded to single-failure-proof, will be operated in tandem as a single-failure-proof lifting device to provide the required lifting capacity. The rated capacity of the two cranes operating in tandem will be 300 tons. The licensee evaluated a number of options to move the heavy loads associated with the repair project through the auxiliary building. To accomplish this task for heavy loads, excluding the steam generator lower assemblies, the licensee has decided to procure a 150-ton capacity overhead bridge crane that meets the single-failure-proof criteria of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" and the applicable sections of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

8.1 Modified Auxiliary Building Crane

By letter dated April 10, 1987, the licensee requested NRC approval for the modification of their current auxiliary building crane as a single-failure-proof crane. The existing crane was not built to single-failure-proof criteria but was found by the NRC staff to satisfy the criteria of NUREG-0612 and Technical Specification Section 3.9.7. However, the licensee has designed a new trolley that meets the single-failure-proof criteria identified in NUREG-0554 and is modifying the auxiliary building structure under the requirements of 10 CFR 50.59. The structural modifications to the auxiliary building and the procurement of a second crane have been required to accomplish the steam generator repairs. The design-rated load for this crane is 150 tons, while the maximum critical load (the load during the Safe Shutdown Earthquake (SSE) event) is 55 tons.

8.1.1 Evaluation for Single-Failure-Proof Requirements

The licensee has addressed the criteria of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The NRC staff reviewed and issued several questions with regard to the licensee's April 10, 1987 submittal. The licensee has addressed the staff concerns in the July 20, 1987 submittal. Therefore, the staff finds that the licensee has demonstrated that the modified crane meets the single-failure-proof criteria of NUREG-0554.

8.1.2 Seismic Evaluation

The licensee has provided the results of the seismic evaluation of the modified crane as documented in the original submittal of April 10, 1987 and additional submittal of October 6, 1987. This evaluation was performed by the original crane supplier, Whiting Corporation. The modified 150-ton single-failure-proof crane has been qualified by Whiting Corporation to a maximum critical load of 55 tons, considering the main hook capacity with the trolley situated at mid-span and the hook in the down position.

The crane was modeled as a multi-degree of freedom system. The "ANSYS" computer code was used for the static and dynamic analyses. The dynamic model analyses provided the natural frequencies, expanded mode shapes, and resulting forces and stresses. The amplified response spectra considered three orthogonal excitation for the specified earthquakes. Also, all modes were combined by the method specified by Regulatory Guide 1.92, Revision 1. The seismic response results were combined by the SRSS method and then absolutely added to the results from the static load effects. The analyses assure that the reaction forces along the trolley wheels are evaluated for the maximum value that the system could sustain before any slippage would occur.

The crane is mainly constructed of ASTM A 36 structural steel with a minimum yield strength of 36 Ksi for operating basis earthquake (OBE) and 32.7 Ksi for safe shutdown earthquake (SSE). The resulting maximum stresses occurred in the crane girder and their values are 22.4 Ksi for OBE and 31.8 Ksi for the SSE conditions. The licensee also evaluated the crane's rope loads. The maximum rope corresponding loads are 363 Kips for OBE and 577 Kips for SSE conditions as compared to their allowable loads of 1200 Kips and 1630 Kips, respectively.

Since all of the above results are less than their corresponding allowable values, they are acceptable by the staff.

During a meeting with the licensee on September 15, 1987, the NRC requested that the licensee evaluate the existing auxiliary building to demonstrate that at a controlling structural location the new loads to be imposed by the modified crane would not result in overstressed structural components. The licensee has provided the results of this evaluation in their submittal dated December 4, 1987, including a comparison of resulting stresses for the building girders that support the cranes and trolleys, the building support columns, and the reinforced concrete elements between the interior support columns. The resulting stresses are below the allowable material values for the conditions imposed by two 150-ton single-failure-proof cranes lifting the steam generator lower assembly and by the combined 115-ton load during the safe shutdown earthquake. The staff finds the results of this evaluation acceptable.

8.1.3 Conclusion

The licensee has addressed the staff's concerns raised and highlighted above. The licensee utilizes a computer code that is found in the public domain and whose analysis results are below the allowable values identified by design codes acceptable to the staff. Also, the licensee followed the criteria for single-failure-proof cranes identified in NUREG-0554 in designing the crane modifications. Based on the above evaluation findings, the staff concludes that the proposed modifications are acceptable.

8.2 Evaluation of Modification to the Auxiliary Building

The NRC staff has reviewed the licensee's proposed modifications to the auxiliary building, including the crane girders, two end column supports for the crane girders, and the spent fuel pool crane, as follows:

a. Crane girder webs at and near each vertical support.

b. Main bearing stiffner at each main column support.

c. Web splicing bolting at two bolted field splices on each girder.

d. Crane rail splices.

e. Reinforcing at corner of girder at T-Line support.

f. Reinforcing at Columns at T-Line.

g. Local welding reinforcement at specified locations.

h. Horizontal connections to resist forces acting perpendicular to the crane runway.

i. Horizontal connections to resist forces acting parallel to the runway girders.

j. Addition of seismically qualified platform on top of the fuel handling bridge crane.

The above modifications have been identified in AEP Drawings Nos. S-SGR-12-3127-1, S-SGR-12-3128-0, and S-SGR-12-3129-0. The licensee has performed hand and computer calculations evaluating these modifications. The calculation summary tables presented by the licensee indicate that the capacity of the affected structural components after the modification would exceed the capacity demands resulting from future loading conditions. The future loading conditions include those resulting from operation of the modified cranes during normal plant operation and during the steam generator modifications.

Based on our discussions with the licensee and the review of the results of the analytical computations, the staff has concluded that the licensee has utilized proper and acceptable approaches for the evaluation of the proposed modifications. Therefore, the proposed modifications in the auxiliary building are acceptable.

8.3 New Single-Failure-Proof Crane

The licensee has requested acceptance by the NRC staff of installation of a new 150-ton single-failure-proof crane that will be used in conjunction with an existing modified single-failure-proof crane to accomplish the repair program for the steam generators. Also, to avoid any possible overstress of structural components of the auxiliary building, several modifications have been implemented under the requirements of 10 CFR 50.59.

The licensee has addressed the guidelines of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The staff has reviewed the licensee's commitments in parallel with the review of the modification of the existing crane into a single-failure-proof crane. The staff has issued several questions to the licensee to clarify their commitments to NUREG-0554. The licensee has addressed the staff concerns for both the modified and the new crane in subsequent submittals.

The new crane differs slightly from the modified crane in that it is qualified at 60-ton maximum critical load capacity during the postulated SSE event. The new and the modified cranes have been analyzed by Whiting Corporation. The new crane is qualified for a higher seismic load (60 tons vs. 55 tons for the modified one) mainly due to a difference in the crane bridge. The crane seismic evaluation is contained in Whiting Corporation's Report C 6766, Vols. 1 & 2, dated September 9, 1987.

The crane was modeled as a multi-degree of freedom system with node points interconnected by appropriate finite elements. "ANSYS", a public domain computer program, was used to perform the modal and static analyses. The design required that the excitation along the crane runway be proportioned to prevent slip of the crane system. The results of the seismic evaluation have indicated that the required structural components did not exceed the allowable stresses with a 60-ton load on the main crane hook, under SSE conditions.

Based on the licensee's demonstration of compliance to the guidelines of NUREG-0554 and the adequacy of the crane's structural component in meeting their allowable stress values, the staff finds that the proposed new crane installation is acceptable.

Based on the fact that single-failure-proof cranes will be used to move heavy loads through the auxiliary building, it is concluded that handling of heavy loads in the auxiliary building will be accomplished without affecting equipment important to the safe operation of Unit 1 or the spent fuel pool and associated cooling and purification equipment.

8.4 Conclusion

Based on the above evaluation, the staff finds that the movement of heavy loads for the steam generator repair project is acceptable.

9.0 STRUCTURAL COMPONENTS AND FIRE PROTECTION

The steam generator repair will include the complete replacement of the steam generators' lower assemblies. This action necessitates removal of portions of the reinforced concrete doghouse structures that surround the steam generator.

The licensee has documented the details of the concrete removal in Section 3 of their, "Steam Generator Repair Report". The licensee plans to follow all of the applicable commitments in the Donald C. Cook FSAR for the construction and repair program. This will include adherence to NRC and industry standards for the concrete and steel reinforcement, except for mechanical splices. The NRC staff raised a concern on the adequacy of the original proposed procedures for splicing of the reinforcing bars. In response to the staff concern, the licensee revised the original proposal to include additional on-site quality assurance testing and inspection procedures. As modified by their recent submittal of October 20, 1987, the revised mechanical splicing program is found acceptable by the staff. The proposed method of replacement for the portions of each doghouse structure is found acceptable by the staff because the licensee has committed to follow the commitments in the Donald C. Cook FSAR, and to adopt additional quality assurance, testing, and inspection procedures identified in their submittal of October 20, 1987.

The repair program applicable to the removal of portions of the reinforced concrete doghouse surrounding the steam generators is also acceptable to the staff because it follows the commitments of the Donald C. Cook FSAR and adopts additional quality assurance, testing, and inspection to be provided by the licensee during the installation of the required Caldweld's splices. Based on these findings, the staff concludes that the proposed steam generator repair program, in relationship to the removal and reinstallation of structural components, is acceptable.

The fire barrier evaluations for the affected fire areas were reviewed to assess the impact of increased fire loadings and fire hazards due to construction activities. Construction activities were determined not to impact the validity of these evaluations with respect to Unit 1 shutdown capability provided there is no continuity of combustibles, such as wooden temporary stairs and trash chutes, between the crane bay and the 650' elevation of the auxiliary building which could promote rapid fire spread. Temporary stairs and other structures connecting these elevations will be made primarily of non-combustible materials or compensatory measures will be provided.

The effect of a Unit 2 construction fire was evaluated by assuming that the equipment in Unit 2 containment and auxiliary building fire areas directly affected by construction activities would be damaged. Loss of all equipment in the postulated fire areas would not cause loss of Unit 1 safe shutdown capability.

10.0 TRANSIENT AND ACCIDENT ANALYSIS

The licensee has made a comparison of the operating and design parameters for the original and the repaired steam generators in Section 2.2 of the repair report. The comparison shows that the repaired steam generators will result in very little change to the original operating parameters. Therefore, the licensee concluded the impact on accident and transient analysis will be insignificant. The licensee had Westinghouse perform accident and transient analysis of all applicable accidents contained in Chapter 14 of the Donald C. Cook FSAR with the parameters of the repaired steam generators. The accidents evaluated included the following:

Non-Loss-of-Coolant Accidents

- Decrease in Feedwater Temperature
- Increase in Feedwater Flow
- Increase in Steam Flow
- Inadvertent Opening of a Steam Generator Relief or Safety Valve
- Steam System Piping Failures
- Loss of External Load
- Turbine Trip
- Loss of Condenser Vacuum
- Loss of Non-Emergency AC Power to Station Auxiliaries
- Loss of Normal Feedwater
- Feedwater System Pipe Breaks
- Loss of Forced Reactor Coolant Flow
- Reactor Coolant Pump Rotor Seizure
- Reactor Coolant Pump Shaft Break
- Uncontrolled Rod Control Cluster Assembly (RCCA) Withdrawal from Subcritical or Low Power Condition
- Uncontrolled RCCA Withdrawal (Power)

- Control Rod Misoperation
- Startup of an Inactive Loop
- Chemical and Volume Control System (CVCS) Malfunction Resulting in Reduced Reactor Coolant System (RCS) Boron Concentration
- Inadvertent Loading of a Fuel Assembly in an Improper Position
- Spectrum of Rod Ejection Accidents
- Inadvertent Operation of ECCS that Increases RCS Inventory
- CVCS Malfunction that Increases RCS Inventory
- Waste Gas System Failure
- Radioactive Liquid Waste System Leak or Failure
- Radioactive Releases Due to Liquid Containing Tank Failure
- Radiological Consequences of Fuel Handling Accident
- Spent Fuel Cask Drop Accident
- Inadvertent Opening of a Pressurizer Pressure Relief Valve
- Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

Loss-of-Coolant Accidents (LOCA)

- Large Break LOCA
- Small Break LOCA
- Steam Generator Tube Rupture

Accidents and transients reviewed by the licensee were similar to those reviewed and approved by the staff in previous steam generator replacement projects at Surry, Turkey Point, Point Beach and H. B. Robinson.

Based on the results of the reanalysis, the licensee concludes that the repair of the steam generators would not result in any adverse changes in the accident conditions used in the original licensing basis of Donald C. Cook Unit 2. Therefore, the accident and transient analysis contained in Chapter 14 of the FSAR is bounding for the repaired steam generators.

Each transient and accident analysis presented in the Steam Generator Repair Report was determined by the licensee to be bounded by the existing Donald C. Cook Nuclear Plant FSAR Transient and Accident Analyses. On the basis of the information discussed above, the staff concludes the existing analysis is acceptable, and the new steam generators have an insignificant impact on the transient and accident analysis.

11.0 PHYSICAL SECURITY ASPECTS

The licensee, by letter dated February 4, 1987, submitted proposed revisions to Chapter 8 of the Donald C. Cook Nuclear Plant Modified Amended Security Plan (MASP) in accordance with the provisions of 10 CFR 50.54(p). The NRC staff transmitted their approval of the changes to Chapter 8 of the MASP by letter dated September 29, 1987. The details of the changes to the MASP are safeguards information and are being withheld from public disclosure.

12.0 POST INSTALLATION TESTING

The repair of the steam generators in the Donald C. Cook Unit 2 will have minimal impact on existing equipment except for the steam generators and their associated piping and instrumentation. Prior to restart of Unit 2, the licensee plans to run the following tests to assure that the plant is returned to safe and reliable full power operation.

- Both the primary and secondary sides of the steam generator will be pressure tested in accordance with applicable codes.

- Thermal expansions of the RCS will be measured to verify that the steam generators can expand and contract without obstruction. Where necessary, clearances will be adjusted to allowable limits.

- Calorithmetric test to verify adequate reactor coolant flows will be conducted in accordance with appropriate Technical Specifications.

- Reinstallation of affected instruments will be verified. Appropriate functionality tests will be performed on affected instruments.

- Restoration of electrical wiring and cables will be verified and appropriate functionality testing performed.

- Thermal performance testing will be conducted as necessary to verify the thermal performance parameters of the repaired steam generators.

- Steam generator water level stability testing will be preformed to verify stability of automatic level control system.

Testing requirements are identified in the Technical Specifications and the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition. The NRC staff has reviewed this proposal and the requirements and finds that it adequately addresses the broad scope of restart testing and as such is sufficient to commence the steam generator repair program. In discussions with the licensee, the staff indicated a concern with the lack of detail in the Steam Generator Repair Report regarding support system startup testing and requested specific information. By letter dated February 19, 1988, the licensee committed to provide the staff with the specifics of the program for testing and return to service of plant system and components following completion of the repair project. Prior to restart of Unit 2, the NRC staff will review and approve the restart testing program.

13.0 TECHNICAL SPECIFICATIONS

For purposes of the repair project, the licensee has requested temporary relief from 14 Technical Specifications that would be applicable in Mode 6. Although there will be no fuel in the reactor vessel, the licensee will meet those Technical Specifications applicable in Mode 6 with the exception of those listed in Table 3.2-3 of the Steam Generator Repair Report dated November 7, 1986, through Revision 6, dated February 18, 1988.

Technical Specifications 3.1.1.3, 3.1.2.1, 3.1.2.5, 3.9.1, 3.9.2, 3.9.8.1, and 3.9.8.2 each ensure that adequate controls are in place to protect the core from transients and accidents. However, with no fuel in the reactor vessel, these core safeguards are not required. Technical Specifications 3.3.3.9 and 3.3.3.10 ensure that appropriate radioactive and gaseous effluent monitors are maintained operable to monitor and control radioactive releases. Since the steam generators will be taken completely out of service, there will be no mechanism to transport any radioactive effluent to the Steam Generator Blowdown Line Monitor, the Steam Generator Blowdown Treatment Effluent, the Condenser Evacuation System Noble Gas Activity Monitor, Condenser Evacuation System Effluent Flow Rate Monitor, Gland Seal Exhaust Noble Gas Activity Monitor, or the System Effluent Flow Rate Monitor. Therefore, there is no need to maintain these monitors operable.

Technical Specification 3.4.7 ensures limitations on RCS chemistry are maintained. The limitations ensure that corrosion of the RCS is minimized and reduces the potential for RCS leakage or failure due to stress corrosion. During approximately six months of the repair project, the RCS will be at half loop, the reactor vessel head will be in place and the Residual Heat Removal Pumps shut down. Therefore, samples of the RCS coolant will not be obtainable.

The licensee's engineering evaluation has determined that the structural integrity of the RCS will not be diminished during the repair project. In addition, subsequent to the repair project (after sampling is reestablished) the RCS chemistry limits will be verified. Should the chemical properties of the as-found RCS coolant be determined to be outside the acceptable Technical Specification limits, appropriate evaluations will be performed.

Technical Specifications 6.5.1.6(a), 6.8.2, and 6.8.3 ensure that appropriate safety reviews, of procedures and changes thereto, are performed. The reviews ensure that procedural activities are evaluated for their impact on nuclear safety. The Steam Generator Repair Report, SG Repair Quality Assurance Program and procedures for return to service testing, identified in Table 3.2-3 of the Repair Report, define the appropriate scope of procedures and documents which should be reviewed by the Plant Nuclear Safety Review Committee. This part of the Repair Report does not represent an exclusion from existing Technical Specification requirements. This part does clarify how the existing Technical Specifications will be met with regard to the steam generator repair project.

Technical Specification 6.12.2 ensures that proper precautious are established to prevent inadvertent exposure of personnel to high radiation. The proposed change to this specification is purely an administrative reassignment of responsibility. The proposed change would place those high radiation areas, turned over to the steam generator repair project team, under the control of the Project Health Physicist instead of the Plant Health Physicist.

Based on the above the NRC staff concurs with the licensee that granting temporary relief from the Technical Specifications addressed in Table 3.2-3 of the Donald C. Cook Steam Generator Repair Report, dated November 7, 1986, through Revision 6 dated February 18, 1988, will not adversely affect the health and safety of the public.

14.0 ENVIRONMENTAL CONSIDERATION

By letter dated November 23, 1987, the staff issued an Environmental Assessment and a Notice of Issuance of Environmental Assessment and Finding of No Significant Impact in response to the March 12, 1987 application. The staff concluded that there are no significant radiological or nonradiological impacts associated with the replacement of the steam generators and the replacement will have no significant impact on the quality of the human environment. Therefore, pursuant to 10 CFR 51.31, an environmental impact statement has not been prepared for the replacement of the steam generators. The Notice of Issuance of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on December 2, 1987 (52 FR 45881).

15.0 CONCLUSION

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

Principal Contributors:
- J. Stang
- F. Rinaldi
- H. Conrad
- J. Minns
- M. Phillips
- T. Vandel
- B. Burgess

Date: 3/8/88

7590-01

U.S. NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 100 to Facility Operating License No. DPR-74, issued to Indiana Michigan Power Company (the licensee), which revised the license for operation of the Donald C. Cook Nuclear Plant, Unit No. 2 (the facility), located in Berrien County, Michigan. The amendment is effective as of the date of issuance.

This amendment approves the steam generator repair program for the facility and provides a license condition related to the repair operation.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings, as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on March 19, 1987 (52 FR 8680). No request for a hearing or petition for leave to intervene was filed following this notice.

Also in connection with this action, the Commission prepared a Notice of Issuance of Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on December 2, 1987, at 52 FR 45880.

For further details with respect to this action, see (1) the application for amendment dated March 12, 1987, and the licensee's Steam Generator Repair Report submitted by letter dated November 7, 1986, as revised March 30, July 24, October 20 and December 4, 1987, and February 18, 1988, (2) Amendment No. 100 to License No. DPR-74, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Maude Preston Palenski Memorial Library, 500 Market Street, St. Joseph, Michigan 49085. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 8thday of March 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Wayne E. Scott Jr

Wayne E. Scott, Jr., Acting Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects