DCS MS-016

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Docket No. 50-266

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Mr. C. W. Fay
Vice President - Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant Unit No. 1 in response to your application dated May 27, 1982 as supplemented by letters dated July 22, 1982, August 9, 1982 and March 1, 1983.

The amendment approves the steam generator repair program for the Point Beach Nuclear Plant Unit 1 and requires as a license condition that the repair operations be conducted in accordance with licensee commitments identified in the approved Point Beach Nuclear Plant Unit No. 1 Steam Generator Repair Report dated August 9, 1982 and revised March 1, 1983 and additional commitments reflected in the staff's Safety Evaluation (SE).

Minor changes to the steam generator replacement activities other than those described in the Steam Generator Repair Report Revision 1 are authorized where it can be demonstrated that the changes are:

(1) bounded by the considerations described in the staff's Safety Evaluation and

(2) do not change commitments described in the Repair Report and in the staff's Safety Evaluation.

Where such changes are necessary, we request that you inform us in writing of these changes and have the appropriate documentation available for review justifying that the changes meet conditions 1 and 2 described above. Where proposed changes cannot be demonstrated to meet the conditions described above, prior NRC review and approval are necessary prior to return to power operation.

The licensing action was noticed in the Federal Register on July 12, 1982 (47 FR 30125) and our intent to notice was transmitted to you in our July 6, 1982 letter. Subsequently Wisconsin's Environmental Decade (WED) filed a Petition for Leave to Intervene and request for hearing by letter dated August 10, 1982.

Following a Special Prehearing Conference held on November 19, 1982 on this matter, the Atomic Safety and Licensing Board (ASLB) dismissed WED's Petition for Leave to Intervene by Order dated December 10, 1982. The ASLB cited WED's failure to proffer one good contention with adequate bases and WED's willful absence from the Special Prehearing Conference as grounds for dismissal.

Though appealed by WED, the ASLB Order has been upheld by the Atomic Safety and Licensing Appeal Board (ASLAB) in their March 22, 1983 Decision.

In the December 10, 1982 Order, the ASLB suggested the NRC staff evaluate additional information discussed during the Special Prehearing Conference. We have obtained that information from your staff and our evaluation of that additional information is included in the Safety Evaluation supporting this amendment. Further additional information was requested from the licensee by the ASLAB in a March 22, 1983 Order. After reviewing the licensee's response, the ASLAB requested in a July 8, 1983 Order that the staff evaluate the efficacy of eddy current testing to detect flaws in steam generator tubes. The staff responded to this request in the "Affidavit of Herbert F. Conrad" dated July 28, 1983. In a subsequent Order dated September 7, 1983 the ASLAB requested clarification to certain portions of Mr. Conrad's affidavit. The staff plans to issue a response by mid-October 1983.

By letter dated July 6, 1983 the NRC staff informed you of its intention to issue an Environmental Impact Statement with regard to the proposed repair at Point Beach Unit 1. In that letter we requested additional information in order to complete our review. You responded to our request by letter dated July 7, 1983. Our Draft Environmental Statement (DES) was issued July 15, 1983 with Notice of Issuance provided in the Federal Register on July 22, 1983 (48 FR 33531). The comment period expired on September 6, 1983. Our Final Environmental Statement with respect to this repair was issued on September 30, 1983.

Originally, the staff's SE was issued on July 15, 1983. However, slight changes to the SE were made in order to more accurately reflect the location of the licensee's commitments with regard to Quality Assurance and in order to address the Department of Health and Human Services comments to the DES regarding breakdown of estimated dose by job category. Therefore, the revised SE is being reissued with this amendment.

A copy of our Notice of Issuance is also enclosed.

Sincerely,

Original Signed by J. R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 75 to DPR-24
2. Safety Evaluation
3. Notice of Issuance

cc: See next page

ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD
PMKreutzer	TColburn/pn	JRMiller	GCLainas	M. KARMAN
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

DISTRIBUTION:
Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-266

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: WISCONSIN ELECTRIC POWER COMPANY, Point Beach Nuclear Plant, Unit No. 1

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (6) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 75.
  Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#3:DL					
SURNAME	PMKreutzer/pn					
DATE	10/3/83					

NRC FORM 102 7—79

Wisconsin Electric Power Company

cc:

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Ms. Kathleen M. Falk
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Wisconsin's Environmental Decade
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated May 27, 1982 as supplemented by letters dated July 22, 1982, August 9, 1982 and March 1, 1983 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 Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-24 is amended by adding paragraph 3.J. to read as follows:

   J. The licensee is authorized to repair Unit 1 steam generators by replacement of major components. Repairs shall be conducted in accordance with the licensee's commitments identified in the Commission approved Point Beach Nuclear Plant Unit No. 1 Steam Generator Repair Report dated August 9, 1982 and revised March 1, 1983 and additional commitments identified in the staff's related Safety Evaluation.

3. This license amendment is effective immediately upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Date of Issuance: September 30, 1983

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-266

1.0 INTRODUCTION

Wisconsin Electric Power Company (licensee or WE), by letter dated May 27, 1982, notified the NRC of their intent to repair the two Point Beach Unit 1 steam generators by replacement of major components, including tube bundles. The licensee indicated that they had reviewed the repair process pursuant to 10 CFR 50.59 and that they had concluded that prior NRC approval was not required because the repair program did not require a change to Unit 1 Technical Specifications, did not involve an unreviewed safety question and did not present a significant hazards consideration. The licensee stated that, if the NRC staff disagreed with their determination, the staff should consider the May 27, 1982 letter as an application for amendment to the Point Beach Unit 1 license.

By letter dated July 6, 1982 the NRC staff informed the licensee that prior NRC approval was required for this repair action, that the staff was considering the licensee's May 27, 1982 letter as an application for amendment to the Point Beach Unit 1 license and that this licensing action was being noticed in the Federal Register in order to provide opportunity for hearing. Notice of Consideration of Issuance of Amendment to Facility Operating License in connection with this action was published in the Federal Register on July 12, 1982 (47 FR 30125).

By letter dated August 10, 1982 Wisconsin's Environmental Decade (WED) filed a Petition For Leave to Intervene and requested an opportunity for hearing in this matter. An Atomic Safety and Licensing Board (ASLB) was established on August 18, 1982 to rule on petitions for leave to intervene and/or requests for hearing and to preside over the proceeding in the event that a hearing was ordered.

A Special Prehearing Conference was scheduled for November 19, 1982. WED failed to attend that Special Prehearing Conference and by Order dated December 10, 1982, the ASLB ordered that WED's August 10, 1982 Petition for Leave to Intervene be dismissed, citing WED's failure to attend the prehearing conference and WED's inability to provide at least one good contention with adequate basis as independent and separate reasons for dismissal.

WED subsequently appealed the ASLB dismissal Order by letter dated December 20, 1982. The Atomic Safety and Licensing Appeal Board upheld the ASLB Order of December 10, 1982 in a Decision issued on March 22, 1983.

On April 7, 1983 WED filed a Petition for Review of Appeal Board Decision pursuant to 10 CFR 2.786(b) requesting that the Commission undertake review of the Atomic Safety and Licensing Appeal Board's decision.

The purpose of this Safety Evaluation is to document the results of the NRC staff's review of the safety significance of the licensee's proposed steam generator repair for Point Beach Unit 1.

2.0 BACKGROUND

In the past, Point Beach Nuclear Plant Units 1 and 2 have experienced various corrosion problems in their steam generators. The problems include caustic intergranular attack of the tubes in the crevice region of the tubesheet and phosphate wastage thinning above and usually within 2 inches of the top of the tubesheet. These problems have been more severe for Unit 1 than Unit 2 and resulted in the NRC issuing Orders for Modification of License for Unit 1 dated November 30, 1979 as modified by Orders dated January 3, 1980 and April 4, 1980. These orders imposed, among other things, more frequent eddy current inspections, more restrictive reactor coolant radioactivity levels, much more restrictive steam generator tube leakage rates and operation at reduced primary pressure for Unit 1.

In an effort to find an acceptable fix to the steam generator tube corrosion problem, WE has submitted an application dated May 27, 1982 modified July 22, 1982, for a license amendment which would allow them to repair steam generators for Unit 1. In support of this requested change, the licensee has filed with the NRC staff for its review a Westinghouse Steam Generator Report dated August 9, 1982 revised March 1, 1983 containing technical information regarding steam generator replacement of the Point Beach Unit 1 steam generators.

3.0 SCOPE OF WORK TO BE PERFORMED

The steam generator repairs will be essentially identical to the steam generator repairs conducted at the Surry Power Station Units 1 and 2 and similar to those conducted at Turkey Point Nuclear Generating Station Units 3 and 4.

3.1 Removal and Reinstallation Operations

The repair will consist of replacing the lower assembly of each steam generator including the shell and the tube bundle and refurbishing and partially replacing the moisture separation equipment in the upper assembly.

The old lower assembly will be removed from the containment building through the existing equipment hatch and transported to a special storage facility. The new lower assemblies will arrive at the site by barge or rail. They will be transferred to a wheeled transporter and hauled to the containment building equipment hatch. The old lower assemblies will be sealed prior to transport.

Prior to the repair work, the unit will be shut down and all systems will be placed in a condition for long term layup. The reactor vessel head will be removed for refueling. All of the normal procedures for fuel cooling and fuel removal will be followed. The fuel will be removed from the reactor and placed in the spent fuel storage facility. The reactor vessel head will be replaced. The equipment hatch will be opened and access control will be established. Guide rails will be installed for transporting the lower assembly through the equipment hatch.

After this preparatory work, the cutting of system piping can begin. This will include cutting and removal of sections of steam lines, feedwater lines, reactor coolant inlet and outlet lines, and miscellaneous smaller lines for the service air and water and the instrumentation system. The steam generator supports will be disassembled and the steam generator lower assembly will be lowered and placed in a horizontal position on a transport mechanism. This mechanism will carry the assembly through the equipment hatch. A mobile crane will lift the lower assembly onto a transporter that will carry it to the steam generator storage facility on the site.

After removal and storage of the steam generator lower assemblies, their replacement will be transported from the barge dock or temporary storage location to the equipment hatch. The same machinery used to remove the lower assemblies will be used to install the new assemblies in their cubicles. The steam generator support system will be reinstalled and the upper assembly with its refurbished internals will be mounted on the lower assembly. After welding the two assemblies together, they will be stress relieved and inspected in accordance with the ASME Boiler and Pressure Vessel Code (ASME Code). The reactor coolant piping will then be reinstalled in a manner similar to that used during the original installation.

Before startup there will be cleaning, hydrostatic testing, baseline inservice inspections, and pre-operational testing of instruments, components and systems. Then the reactor will be refueled and startup tests will be performed. The performance of the repaired steam generators will be tested for moisture carryover and verification of thermal and hydraulic characteristics.

3.2 Post Installation Testing

A detailed preoperational testing program will be carried out by WE prior to fuel loading to reestablish the integrity of the reactor coolant system and the main steam and feedwater system, to ensure that all systems are in operating condition and to provide baseline data for future performance testing.

Following the completion of the major installation activities, the unit will be restored to a condition for testing and inspection. The following activities will be performed: hydrostatic tests in accordance with Section XI of the ASME Code and baseline inservice inspection on piping, equipment and components, including 100 percent eddy current inspections of steam generator tubing.

After the residual heat removal system has been tested and placed in service, fuel will be transferred to the reactor vessel. One third of the fuel assemblies placed in the vessel will be new fuel assemblies and the operation will not differ significantly from a normal refueling.

During the initial startup of the unit, tests will be performed to verify the thermal and hydraulic performance of the nuclear steam supply system.

4.0 FABRICATION, STRUCTURAL INTEGRITY, COMPONENT DESIGN MODIFICATION AND CORROSION ASPECTS OF THE REPAIR PROGRAM

Westinghouse Electric Corporation will fabricate new steam generator lower assemblies. The design of the lower assemblies will match the design performance of the lower assemblies being replaced. However, several design features that do not alter mechanical performance and FSAR parameters are included in the design. These design features are intended to provide improved thermal hydraulic performance, improved access to the tube bundle, and reduce the potential for secondary side corrosion. Regulatory Guides applicable to fabrication are listed in Table 1.

Table 1

Applicable Regulatory Guides for Fabrication of the Replacement Lower Assemblies

1. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam and Radioactive-Waste-Containing Components of Nuclear Power Plants" (Rev. 2), July 1975.

2. Regulatory Guide 1.29, "Seismic Design Classification" (Rev. 3), July 1978.

3. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Design of Nuclear Power Plants" (Rev. 1), December 1973.

4. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nulcear Power Plants" (Rev. 1), December 1973.

5. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" (Rev. 0), October 1973.

6. Regulatory Guide 1.68 "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors", August 1978.

7. Regulatory Guide 1.84, "Code Case Acceptability-ASME Section III Design and Fabrication" (Rev. 19), April 1982.

A. Westinghouse controls its suppliers to:

1. Limit the use of code cases to those listed in Regulatory Position C.1 of the applicable guide revision in effect at the time equipment is ordered except as described in item B. below.

2. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, where use of such cases is needed by the supplier.

3. Allow continued use of a case considered acceptable at the time of equipment order, where such code was subsequently annulled or amended.

B. Westinghouse will seek NRC permission for the use of code cases needed by suppliers and not yet endorsed by Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, and permit supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later revision of the regulatory guide includes endorsement).

8. Regulatory Guide 1.92, "Combination of Modes and Spatial Components in Seismic Response Analysis" (Rev. 1), February 1976.

9. Regulatory Guide 1.121 "Bases for Plugging Degraded PWR Steam Generator Tubes", April 1977.

4.1 ASME Code Application

The original steam generators were built to the 1965 Edition of the ASME Boiler and Pressure Vessel Code (ASME Code), including Addenda through Summer 1966; the replacement steam generator lower assemblies will be constructed to the latest edition of the ASME Code in effect as of December 1, 1979. However, the stress analysis will be performed using the 1965 Edition of the ASME Code, including all Addenda through Summer 1966, in order to use the same procedures for those portions of the steam generator which were not replaced.

4.2 Quality Assurance (QA)

The QA program for the steam generator repair for Point Beach Plant Unit No. 1 is described in Sections 2.4 and 3.6 of Revision 1 to the Wisconsin Electric Power Company (WE) report of March 1983, "Point Beach Steam Generator Repair Report." The program is applicable to all safety-related activities to be conducted for the steam generator repair including design, disassembly, removal, fabrication, installation, inspection, and return-to-service testing.

The application of Regulatory Guides for QA is addressed in the references provided in the above noted sections and Section 2.1.4 of the WE report. The

Regulatory Guides listed in Table 2 are considered applicable. The specific revisions for the different activities are a function of the performer's QA program.

The staff has reviewed this information to verify that the QA commitments are acceptable for the proposed steam generator repair work. Our acceptance criterion is (1) that QA program commitments previously found acceptable by the NRC will be applied to the steam generator repair, or (2) that any new or revised commitments are at least as stringent as the existing commitments to assure that the QA program is not degraded.

WE has the overall responsibility for the QA program for the steam generator repair. The WE QA program, described in Section 1.8 of the Point Beach FSAR, is applicable to this work. Similarly, the Westinghouse QA program for its activities is described in WCAPs 8370 and 9245. WE will approve the QA programs of Westinghouse, and Westinghouse will approve the QA programs of its onsite contractors or impose its own QA program on them. WE and Westinghouse will assure implementation of these programs commensurate with the scope of work to verify conformance with the requirements of 10 CFR 50 Appendix B. WE and Westinghouse QA personnel will audit and provide surveillance to assure that all safety-related activities are conducted in accordance with applicable codes, standards, and regulations.

Based on our review and evaluation of Sections 2.1.4, 2.4, and 3.6 of the referenced WE report, the staff concludes that WE has described a QA program which is acceptable for the steam generator repair at Point Beach Nuclear Plant Unit No. 1.

Table 2

Applicable Regulatory Guides for QA

1. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

2. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)."

3. Regulatory Guide 1.29, "Seismic Design Classification."

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

5. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants."

6. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

7. Regulatory Guide 1.58, "Qualification of Nulcear Power Plant Inspection, Examination, and Testing Personnel."

8. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants."

9. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions."

10. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records."

11. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

12. Regulatory Guide 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."

13. Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants."

14. Regulatory Guide 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants."

15. Regulatory Guide 1.146, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants."

4.3 Evaluation of Component Modifications

Several modifications have been made to the steam generators to increase the circulation ratio. The circulation ratio is the total tube bundle flow divided by the feedwater flow and is inversely proportional to the steam quality leaving the tube bundle. As the ratio increases, the lateral flow velocity also increases and thus reduces the number of tubes exposed to low velocity flow and the potential for sludge formation. Low steam quality in the bundle reduces the number of tubes exposed to local steam blanketing and reduces the number of potential areas for chemical impurity concentration.

The physical, thermal and hydraulic characteristics of the steam generators will be at least equivalent to those of the original steam generators. The following additional design features have been incorporated in the design.

1. Flow Distribution Baffle

   A flow distribution baffle has been provided 23 inches above the tubesheet. This baffle has a cut out center section and oversized drilled tube holes. The baffle plate assists in directing flow across the tubesheet then up the center of the bundle through the center cutout. The design is sized to maximize the flow to the center of the bundle and minimize the number of tubes in low-velocity regions. Consistent with this purpose, the design is also intended to cause any sludge to deposit near the blowdown intake where it can be removed. The flow distribution baffle plate material is ferritic stainless steel. Access holes have been provided to allow sludge lancing above and below the baffle plate.

2. Internal Blowdown Design Changes

   Maintenance of the secondary side water chemistry is assisted through the use of the blowdown system. Each steam generator will be designed to have two 2-inch schedule 40 Inconel internal blowdown pipes. The blowdown nozzles on the external portion of the steam generator shall have provisions for connection to 2-½ inch existing blowdown piping. The blowdown intake location is coordinated with the baffle plate design so that the maximum intake is located where the greatest amount of sludge may collect. The modified blowdown system should allow increased capacity blowdown in comparison with the present blowdown arrangement.

Concerns have been raised about the effectiveness of the flow distribution baffle and the improved blowdown system (Items 1 and 2 discussed above). The licensee has provided field, model test and analytical data to provide assurance that these two modifications will result in a reduction of sludge build up.

Replacement steam generators utilizing the flow distribution baffle and the improved blowdown system have been installed in Surry Units 1 and 2 and in Turkey Point Unit 3. Visual examination by fiberoptics and sludge measurement techniques at Surry Unit 2 have shown no significant sludge accumulation after approximately 24 months of operation. Data from Surry Unit 1 and Turkey Point Unit 3 are not available presently. Although field verification of the flow distribution baffle and modified blowdown system is limited, the correlation of sludge buildup on the tubesheet with lateral flow velocity has been verified for steam generators without a flow distribution baffle.

Based on the computer analysis model, CHARM, low flow velocities are predicted off center of the tube lane, i.e., away from the blowdown intake. Furthermore, the measured sludge profile height has been correlated with low tube gap velocities. Therefore, to minimize the number of tubes exposed to a low crossflow velocity, a flow distribution baffle and a modified blowdown system have been incorporated into the Point Beach replacement steam generators.

Based on the computer code analysis, the flow distribution baffle has been designed with the objective of limiting the number of tubes exposed to a sludge settling environment and to limit low crossflow velocities to the center of the tube bundle near the blowdown system.

The correlation of sludge buildup on the tubesheet with lateral flow velocity has also been experimentally verified using a flow visualization model at the Carnegie-Mellon University. The Carnegie-Mellon flow visualization model was composed of 120 tubes in both the hot and cold legs arranged in a 4 x 30 array. The model included a tubesheet, wrapper wall, and a single tube support plate. The model did not include a flow distribution baffle. Sludge particle deposition was simulated using particulate material in a working fluid of Refrigerant 113. This test also confirmed the correlation of measured sludge height with low tube gap velocities.

Based on a review of the field experience with the flow distribution baffle and the improved blowdown system, as well as the analytical and test data discussed above, the staff concludes that these modifications will result in an improved thermal hydraulic performance of the replaced steam generators.

3. One of the changes being made in the repaired steam generators at Point Beach Nuclear Plant Unit 1 is that the tubes are hydraulically expanded to the full depth of the tubesheet holes. Concerns have been raised that the residual stresses left by the hydraulic expansion process create a safety problem. Full depth expansion essentially closes the crevice between the tube and tubesheet thus minimizing the possibility of crevice corrosion.

   There are three different techniques of expanding the tube in the tubesheet:

   (1) Mechanical Rolling. Full depth mechanically rolled steam generators now in operation include two non-domestic plants which have operated for about five years.

   (2) Explosive Process. One such process called WEXTEX has been used at Trojan, Beaver Valley 1, Salem 1, Farley 1 and North Anna 1 and 2. Several years of operating experience has been obtained on these plants.

   (3) Hydraulic Expansion. This technique combines the reduced deformation and low residual stress transition of the WEXTEX

process with the tight sealing of the hard mechanical roll. Hydraulic expansion has been adopted as the optimum process for the current steam generator designs including the replacement lower assemblies for the Point Beach Nuclear Plant. The replacement Surry 2 steam generators are the first operating units with hydraulically expanded tubes. Refinement of the hydraulic process has resulted in expanding all but a small crevice of about one eight inch average depth at the top surface of the tubesheet.

The advantages of the hydraulic expansion process are the reduction of the cold working caused by the mechanical hard rolling and the lower residual stresses at the transition of the expanded to unexpanded region of the tubes. Analyses and experiments have shown these tensile stresses to be of the order of 20 ksi on the OD surface and 20-30 ksi on the ID, which are about half the stresses for a mechanical roll. In addition, the tubes in the innermost eight rows of the bundle will be stress relieved after bending to minimize residual stresses.

4. In addition to the change in material properties resulting from hydraulic expansion, the tubing Inconel 600 material is now specially heat treated to take advantage of the increased corrosion resistance of thermally treated Inconel 600 to stress corrosion cracking (SCC) in both primary and secondary environments. The occurence of SCC, which has been observed in some partially expanded units, is expected to be minimized by the

combination of the full depth hydraulic expansion and the thermal treatment of the Inconnel 600 tubing. Confirmation has been obtained from a number of laboratory tests; the results of several tests are summarized as follows.

(1) To confirm the absence of chemical concentration in the crevice remaining from hydraulically expanding the tube into the tubesheet, a chemical hideout test was performed by the licensee, using a Na_2SO_4 solution as secondary fluid of a model boiler. Testing of a simulated hydraulically expanded tubesheet joint showed no indication of hideout during testing nor was there any precipitate in the tube examination following testing. These observations are contrasted to the results of similar testing on a partially rolled crevice configuration in which a precipitate on the tube surface in the vicinity of the top of the tubesheet was present in the post-test examination, indicating a significant concentration within the tube-tubesheet crevice.

Longer term testing of hydraulically expanded tubes in model boilers continues to confirm that the seal between the tube and tubesheet is adequate and that the upper crevice shows no corrosion. The crevice dimensions in these tests range from 0.102 in. to 0.179 in. depth.

(2) The effect of the residual stresses on the OD surface of hydraulically expanded tubes in the transition region was assessed in a series of tests in two aggressive caustic environments. Thermally treated Inconel 600 tubes were hydraulically expanded into simulated carbon steel and Inconel 600 tubesheets, internally pressurized to 30,000 psi hoop stress (which is well above the operating stress) and exposed on the OD to 10% caustic at both 600°F and 650°F. The behavior of the expanded samples was compared to that of thermally treated tubes which were unexpanded.

In the 600°F test solution, none of the samples, expanded or unexpanded, showed cracking after about one year in test. This test temperature is approximately that of the normal hot leg operating temperature at Point Beach. At the much more aggressive test temperature of 650°F, which is intended to accelerate the corrosion mechanism and provide for more definitive differentiation, unexpanded tube specimens showed cracking in as short as 60 days exposure, whereas the minimum time to leak for hydraulically expanded tubes was 621 days. These tests confirm that hydraulic expansion does not degrade the inherent corrosion resistance of thermally treated Inconel 600 tubes in aggressive caustic environment.

The staff has previously reviewed the licensee's secondary water chemistry monitoring and control program. Based on that review the staff concluded that the program is consistent with the recommendations of the NSSS vendor and the EPRI/SGOG water chemistry guidelines and that the program provides reasonable assurance of inhibiting steam generator corrosion and tube degradation.

The use of full-depth expansion essentially eliminates the tubesheet crevice in which concentration of impurities has occurred in the original steam generators and results in an expansion transition which retains the enhanced corrosion resistance of thermally treated Inconel 600 material. The hydraulic expansion results in substantially lower stresses as compared with the stresses due to cold working caused by the mechanical hard rolling and the residual stresses at the transition of the expanded to unexpanded region of the tubes. The staff, therefore, finds the full-depth hydraulic expansion of the tubes in the tubesheet acceptable with no safety concerns.

5. Offset Feedwater Distribution

Feedwater distribution within the steam generators will be modified so that approximately 80% of the flow is directed to the hot leg side of the bundle and the remaining 20% of the flow is directed to the cold leg side of the bundle. This reduces the steam quality in the hot leg side of the bundle and raises the steam quality in the cold leg side of the bundle. The effect of these changes in steam quality is to shift the point of highest steam quality at the tubesheet elevation toward the center of the bundle. This area is utilized for location of the blowdown intake. Feedwater flow redistribution is accomplished by providing a greater number of flow paths on the portion of the feedwater ring which traverses the hot leg side of the tube bundle.

6. Stainless Steel Support Plates

The support plate material has been selected such that the potential for denting of the tubing due to corrosion in the crevice between the tube and tube support plate is significantly reduced. SA-240 Type 405 ferritic stainless steel has been selected for this application. This material is ASME Code approved and is believed to be resistant to corrosion with the chemistry expected during the operation of the steam generator. In addition, SA-240 has a low wear coefficient when paired with Inconel and has a coefficient of thermal expansion similar to carbon steel. Corrosion of SA-240 results in an oxide which has approximately the same volume as the parent material. In addition to the tube support plates, the baffle plate will be constructed of SA-240 Type 405.

7. Support Plate Design

The quatrefoil tube support plate design consists of four flow lobes and four support lands. The lands provide support to the tube during all operating conditions, while allowing flow around the tube. This design also directs the flow along the tubes which limits steam formation and concentration of impurities at the tube-to-tube support plate intersections. The quatrefoil support plate design results in higher average velocities along the tubes, which should prevent sludge deposition. The combination of higher velocities in the support plate region and corrosion resistant material will minimize the potential for support plate corrosion.

Concerns have been raised about the residual stresses that may exist as a result of the forming process of quatrefoil openings in the support plate. The licensee has provided test data to demonstrate that there is no indication of high residual stresses in this region.

Laboratory tests conducted by Westinghouse utilizing highly stressed Type 405 stainless steel U-bends exposed to caustic and chloride environments, and heated crevice and model boiler tests utilizing actual boiler tests utilizing acutal broached quatrefoil samples exposed to the environments which caused tube denting and cracking of the carbon steel support plates, as well as literature searches, have verified that Type 405 stainless steel, as fabricated, is not susceptible to stress corrosion cracking in the steam generator operating environment.

The fabrication of the Type 405 stainless steel support plates does not produce significant residual stresses. The plate material is initially strengthened by heat treatment and tempered at 1325-1375°F. While the purpose of these heat treatments is to optimize the mechanical properties and corrosion resistance of the material, the tempering operation also minimizes any residual stresses which may be present in the plate material. Small holes are then drilled at the required locations for the quatrefoil openings. The quatrefoil openings are produced by broaching; an operation involving multiple shaving, i.e., progressively removing small amounts of metal by utilizing a tool with stepped cutting edges, which removes less and less metal with each step.

Based on a review of the test data and the fabrication process discussed above, the staff concludes that while the residual stresses caused by the broaching operation have not been analyzed, they are considered to be low based on the results of the test data discussed above. Some general corrosion has been observed at tube support lands in certain accelerated heat transfer tests, but there has been no appearance of stress corrosion cracking indicative of high residual stresses.

Certain modifications and refinements have been incorporated in recent designs to improve thermal hydraulic performances. These are included in the Point Beach design and are discussed below. They do not alter previous safety analyses.

1. Flush Tube to Tubesheet Weld

   The tubes on the replacement lower assemblies will be flush with the tubesheet holes and then welded to the tubesheet cladding. Elimination of the protruding tube stub of the original design results in lower entry pressure losses and, therefore, a lower pressure drop in the primary loop. In addition, a possible point of crud buildup and corrosion is minimized with this design.

2. Moisture Separator Modifications

Since the circulation ratio will be greater in the repaired generators, modifications to the moisture separator equipment will be made to accommodate this increase, and to minimize moisture and soluble corrodent species carryover into the turbines.

The secondary moisture separator external drains will be changed to larger internal drains. The existing primary separator swirl vane barrels will be replaced with a primary moisture separator assembly consisting of one hundred and twelve modular 7" I.D. swirl vane assemblies. These modifications provide improved steam-water separation and reduce moisture carryover.

3. Tube Lane Blocking Device

A portion of the recirculated water exiting at the bottom of the wrapper will tend to preferentially channel to the tube lane and bypass part of the tube array. In order to minimize this tube bundle bypass, a series of plates are installed in the tube lane to block the bypass flow paths. These plates are compatible with sludge lancing.

Operational experience, including necessary maintenance and repair, has led to certain changes in design with the objectives of increasing additional maintainability of the units. These changes are discussed below and do not affect performance or FSAR safety analyses.

1. Access Ports

The replacement lower assemblies are provided with additional access ports. Four six-inch access ports will be located slightly above the tubesheet, approximately 90 degrees apart, with two located on the tube lane below the baffle plate. Two six-inch access ports will be located on the tube lane, between the baffle plate and the first tube support plate. The addition of these access ports should promote inspection of the tubesheet and flow distribution baffle plate.

2. Inspection Ports

One three-inch inspection port is located on the lower shell transition cone at an elevation slightly above the top tube support plate of the tube bundle. This port is located on the tube lane centerline and provides for inspection of the top support plate and the tubing U-bend area.

3. Wet Layup Nozzle

A two inch nozzle is to be added to the upper shell to facilitate the wet layup of the steam generators during periods of inactivity. The wet layup nozzle can be used during these periods to maintain desired water chemistry in the steam generator. The nozzle can also be used in conjunction with other equipment to circulate water through the steam generator during periods of layup.

4. Primary Shell Drain

A 3/8 inch primary shell drain is included in the channel head to provide additional drainage of the channel head. This drain facilitates maintenance and inspection to be conducted in the channel head.

5. Primary Closure Rings

   Closure rings will be welded inside the channel head at the base of each primary nozzle so that closure plates can be installed during primary chamber maintenance. This design allows the plates to be bolted to the rings for quick installation and removal. Closure plates allow maintenance or inspection to be conducted in the channel head with the reactor cavity flooded and, thus, can reduce outage time.

6. Steam Nozzle Flow Limiting Device

   A flow limiting device will be installed in the steam outlet nozzle to minimize the pressure drop across internal components during a postulated steam line break transient and also to help minimize the blowdown rate for the postulated accident condition.

4.4 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. Recent inspections have found a variety of foreign objects in the secondary side of steam generators. Procedures will be implemented by WE to preclude the introduction of foreign objects into the steam generators during the repair. These procedures include a combination of physical barriers and administrative controls. Physical barriers will be specified as part of the Control Work Packages, consistent with the work to be performed. Physical barriers such as herculite, metal plates, and decking will be used in work areas, as appropriate.

Administrative procedures will include procedures for personnel access control, tool control and log-in and log-out procedures. The administrative controls will include such requirements as lanyards on small equipment and personal items, and design features such as lock wires on equipment to prevent loss of material in the steam generators. Following the repair, final inspection and search will be performed on steam generator secondary side. This search will be performed by inserting a fiberscope through the steam generator handholes and conducting a 360 degree search of the annulus at the tubesheet. Any foreign objects which are judged to have the potential for steam generator tube damage, and which are accessible, will be removed.

Surveillance of steam generators during subsequent operation could include periodic inspections of the tube bundles using fibre optic or television techniques during refueling shutdowns, continuous monitoring via loose parts monitoring systems, or a combination of these. Recommendations for loose parts surveillance are currently being developed by the NRC staff. WE has stated that loose parts surveillance programs during subsequent operation will be developed following the NRC recommendations.

4.5 Additional Corrosion-Related Aspects of Repair

In addition to reviewing the design changes of the new lower steam generator assemblies, we have reviewed the following corrosion related areas of concern: 1. the decontamination process used in preparing surfaces for joining or repair; 2. lay-up conditions for portions of the secondary and primary systems to be reactivated; 3. the criteria for the startup condition of metal surfaces in the reassembled secondary water system; and 4. the storage conditions of the degraded and removed portions of the steam generators.

The decontamination of the steam generator and secondary system surfaces that are to be joined or repaired will be accomplished using techniques which are the same as those used during normal plant operations and repair maintenance. The normal procedures require flushing the surfaces with water and then wiping with an absorbent cloth. If any further decontamination of the surface is necessary, an abrasive method will be used with boric acid crystals in a water slurry. This procedure will not introduce materials that are deleterious to the reactor coolant system or the secondary water system.

The parts of the primary system being refurbished or replaced will be kept in a wet layup condition using borated demineralized water with hydrazine addition and/or nitrogen blanketing as appropriate to prevent the intrusion of oxygen. This corrosion control in storage should be adequate to prevent corrosion degradation of primary system surfaces. Layup of secondary systems such as the feedwater system and condenser will be dry. These systems will be drained and dried with air in accordance with normal plant maintenance practice. Careful draining and drying of these secondary systems should be adequate to prevent corrosion degradation during system layup.

The steam generators are sealed and protected against moisture during shipment. During installation the steam generator is open to the normal containment atmosphere but no fluids are allowed within the assembly. Thus, there is assurance the steam generator surfaces are acceptably clean and not degraded.

The lower assembly removal procedures require that the assembly be drained and sealed prior to movement out of the containment. Sealing will be accomplished by welding 3-inch thick steel closure plates over the top of the lower assembly at the girth cut location, over the inlet and outlet reactor coolant nozzles and all other vessel penetrations. This procedure is adequate to prevent the leakage of radioactive material to the outside environment due to internal corrosion during the period of on-site storage of the replaced lower assemblies.

Based on the above evaluation, the staff concludes that: the procedures and controls to be used in the repair/replacement program are adequate to reduce the potential for corrosion degradation of the reassembled primary and secondary coolant systems during layup and subsequent operation; and there is reasonable assurance that the removed portions of the steam generators will not degrade by corrosion during long term storage and affect the public health and safety.

The staff further concludes that the steam generator repair/replacement program is acceptable from the corrosion aspect and that there is reasonable assurance that the public health and safety will not be endangered.

4.6 Conclusions

The following modifications have been made in the replacement steam generators.

1. Flow distribution baffle
2. Internal blowdown piping.
3. Full depth expansion in the tube sheet.
4. Offset feedwater distribution.
5. Thermally treated Inconel 600 tubing.
6. Stainless steel support plates.
7. Support plates with quatrefoil openings.
8. Flush tube to tubesheet weld.
9. Moisture separator modifications.
10. Tube lane blocking device.
11. Access ports.
12. Inspection ports.
13. Wet layup nozzle.
14. Primary shell drain.
15. Primary closure rings.
16. Steam nozzle flow limiting device.

The staff has reviewed these design modifications and evaluated their impact on steam generator performance. Based on our review, we conclude that these modifications meet the requirements of the ASME Codes as required by the Standard Review Plan and will result in improved thermal hydraulic, corrosion resistance and maintainability characteristics. It is further concluded that the repair work and subsequent operation can be conducted without undue risk to the health and safety of the general public and does not involve an unreviewed safety question.

5.0 HANDLING OF HEAVY LOADS

The NRC staff has reviewed the Point Beach Nuclear Plant Unit 1 Steam Generator Repair Report for the handling of heavy loads as follows:

The existing polar crane will be utilized for handling heavy loads 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 is not applicable. Further, any consequences from a load drop would be of an economic nature and not a radiological safety concern.

In response to our concern for assuring safe heavy load handling by the polar crane once normal plant operation begins after completion of the repair program, the licensee committed to the following in a letter dated November 22, 1982:

a. Special heavy lifts during the steam generator repair project will meet the guidelines of ANSI B30.2.0-1976, Section 2-3.2.1, concerning handling of special heavy lifts. Records of special heavy lifts will be maintained in project records as part of the steam generator repair program.

b. The polar crane will be inspected in accordance with ANSI B30.2.0-1976, Section 2-2 guidelines after completion of the steam generator repair project, and prior to returning the crane to normal service. Load testing of the existing polar crane trolley will not be required, as it will not be used for special heavy lifts during the repair, and will not be modified. Further, no structural modifications are planned for the bridge, and thus, a load test for the bridge will not be required. However, should the bridge require modifications, the special heavy lifts performed during the repair would serve as an adequate load test of the modified bridge.

We have reviewed the licensee's response and conclude that the above commitments adequately address our concern and are therefore, acceptable.

6.0 RETURN TO SERVICE TESTING

The licensee indicates that the main feedring will be replaced as part of the steam generator repair program. However, the licensee did not discuss differences between the present and new feedring design or whether operation of the feedwater system will remain the same in order for us to evaluate the potential for unacceptable steam generator water hammer and the possible need for a water hammer test as part of the return to service testing. We note that Point Beach has not experienced damaging steam generator water hammer in the past. In response to this concern, by letter dated November 22, 1982, the licensee further described the feedring modifications and water hammer prevention features in the new steam generator design. The feedring will include top discharge J-nozzles and a welded feedwater nozzle thermal liner to help limit drainage of the feedring and prevent entry of steam during hot standby when the steam generator water level falls below the elevation of the feedring. The feedwater line also includes a check valve located close to the feedwater nozzle to further reduce the potential for significant steam entry into the feedwater line. The licensee also notes satisfactory operating experience with similar steam generators recently installed at Turkey Point Unit 3 and 4 and Surry Units 1 and 2. The licensee maintains the above design features and experience make the performance of a water hammer test unnecessary.

We have reviewed the above response and conclude that it does not sufficiently resolve our concern. It is our position that a further verification is necessary in order to show that no unanticipated problem caused by water hammer would result when the new steam generators are in service and an auxiliary feedwater system demand occurs. Therefore, as discussed with and agreed to by the licensee, the licensee will verify that no water hammer occurs as part of the normal startup testing when auxiliary feedwater is being supplied to the new steam generators. Performance of this verification will resolve our concern in this area.

In all other respects, the licensee's proposed return to service testing program is acceptable.

7.0 ALARA CONSIDERATIONS

WE has committed to making every reasonable effort to maintain occupational radiation exposures as low as is reasonably achievable (ALARA) in accordance with 10 CFR Part 20.1(c) and Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensure that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable." ALARA activities specifically directed to reduce occupational radiation dose to workers include: local decontamination of work areas, personnel training in full-size mock-ups, installation of temporary shielding, use of tents or glove-boxes with air filtration equipment when contaminated piping systems are cut, and the use of automatic welding equipment for welding the reactor coolant piping.

WE has administratively divided their steam generator replacement activities into discrete "work packages." The licensee has committed that each work package will contain appropriate procedures, instructions and drawings to assure that the work can be completed with a minimum of radiation exposure.

The licensee has committed to having an engineer knowledgeable in health physics practices review the work packages for ALARA consideration. In addition, during the actual work, a full-time Health Physics Director will implement all health physics activities, through health physics shift coordinator and health physics technicians.

WE has described a program to maintain occupational doses ALARA during the repair work that is in accordance with 10 CFR 20.1(c) and the guidelines of Regulatory Guide 8.8 and therefore is acceptable.

The staff has reviewed the occupational radiation protection aspects of the Wisconsin Electric Power Company's (WE) proposed steam generator repair program for Point Beach Nuclear Plant Unit 1. The occupational external dose for this program is estimated by the licensee to be approximately 1400 person-rems. This estimate is based on the licensee's breakdown of occupational dose for each phase of the steam generator repair. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job and the average dose rate in the area where the job will be performed. WE's dose estimate for the Point Beach Nuclear Plant Unit 1 steam generator repair is comparable to those of other licensees who perform similar repairs and is therefore acceptable. Attachment 1 shows the estimated dose in person-rems and estimated person-hours associated with each repair activity.

WE has reviewed the potential internal exposure to workers during the steam generator repair activities. The potential for airborne contamination will increase because of the work required during the removal of the steam generator, e.g. cutting into radioactivity contaminated piping. The licensee has committed to following the guidance of Regulatory Guide 8.15 and NUREG-0041 in implementing the respiratory protection program as well as providing special control measures such as use of temporary enclosures and filtered ventilation systems. WE's internal exposure program is in accordance with Regulatory Guide 8.8 and therefore is acceptable.

8.0 TRANSIENT AND ACCIDENT ANALYSIS

The purpose of this section of the Safety Evaluation is to present our evaluation of the Point Beach Unit 1 safety analyses presented in reference 2 for operation of this unit with replaced steam generator lower assemblies. The latter consist of the lower shell, transition cone, the tube bundle, wrapper and other internals. Minor differences between the original and replacement assemblies, which could have a small effect on the safety analysis, include a 2% increase in secondary volume and a 2% decrease in heat transfer surface. There will also be a decrease in steam generator tube pressure drop. Another design modification involves installation of the steam line flow limiter in the steam generator outlet nozzle instead of inside the main steam line.

Point Beach Unit 1 is now operating at reduced pressure, temperature and power because of extensive tube plugging (about 14% at present). The NRC Confirmatory Order of November 30, 1979, as amended on January 5, 1980, and April 29, 1980 (References 3, 4 and 5) allows up to 18% tube plugging and restricts Unit 1 operation to 2000 psia. Replacement of the steam generator lower assemblies will enable operation of Point Beach Unit 1 at either rated primary pressure (2250 psia) or at reduced pressure (2000 psia). The licensee has indicated a preference for operating at 2000 psia because of the beneficial effect of lower operating pressure on NSSS components. A transient analysis for operation at reduced pressure is provided in reference 6.

This section provides our evaluation of transients and accidents that could be significantly affected by the proposed replacement with operation at either 2250 or 2000 psia. These include loss of normal feedwater, loss of AC to auxiliaries, locked rotor, steam line break and LOCA. Differences between the analyses at 2250 psia and 2000 psia are discussed where pertinent. The following transients are not adversely or significantly affected by the proposed replacement at either 2250 or 2000 psia and are therefore not further discussed: Loss of reactor coolant flow, CVCS malfunction, excessive load increase, startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, loss of load, and steam generator tube rupture.

Evaluation of Transients and Accidents

1. Loss of Normal Feedwater

   The FSAR analysis for the loss of normal feedwater transient assumed this event to occur at 102% power, at minimum normal steam generator level, and loss of the reactor coolant pumps. The reactor trips on low-low steam generator level. One auxiliary feedwater pump starts one minute after the low-low steam generator level signal, delivering flow to one steam generator. Secondary steam relief is via the steam generator safety valves. The tube sheet of the steam generator receiving auxiliary feedwater flow is always covered. The capacity of one auxiliary feedwater pump is sufficient to prevent water relief from the primary relief and safety valve. For operation at rated conditions the peak T_{ave} is 609^{o}F at about 1/2 hour after transient start.

With regard to operation at reduced pressure, Reference 6 indicates that, while volumetric expansion of primary coolant during the transient would be somewhat greater, this would be offset by the fact that the initial programmed pressurizer volume is less at reduced pressure. Thus, the conclusions in the FSAR would still be generally valid.

Reference 2 indicates that operation with the replacement steam generator would result in a slight improvement in system behavior for this transient because of the increased secondary mass at full load. The conclusion that the tube sheet of one steam generator will always be covered remains valid. We conclude that the "loss of normal feedwater" analysis is acceptable.

2. Loss of AC Power to Station Auxiliaries

The FSAR analysis for the "loss of A.C. power to station auxiliaries" transient utilizes the same assumptions as for the loss of normal feedwater transient with the exception that two auxiliary feedwater pumps are assumed to deliver flow to both steam generators. The resulting peak T_{ave} is 594°F. The effect of operation at reduced pressure is similar to the "loss of main feedwater" transient. The effect of the replacement steam generator is negligible. We conclude that the "loss of A.C. power to station auxiliaries" analysis is acceptable.

3. Locked Rotor

The FSAR analysis for the locked rotor accident assumes that seizure of one reactor coolant pump (RCP) shaft occurs at 102% power. Reactor trip occurs on a low flow signal. Upon reactor trip, it is assumed that the most reactive RCCA is stuck in its fully withdrawn position. The time from pump seizure to initiation of control rod motion was assumed to be 0.9 seconds. The licensee has stated that test data indicates measured times of 0.45 seconds from the time the low flow trip setting is reached until the instant the rods are released. Another 0.1 seconds is assumed for the interval between pump seizure and reaching the low flow trip set point, for a total "most probable time delay" of 0.55 seconds. Thus 0.9 seconds appears conservative, (Ref. 9). No credit was taken for the pressurizer relief valves, pressurizer spray and steam dump. The licensee assumed offsite power to be available and continued operation of one RCP. This is further discussed below.

The FSAR analysis showed the peak pressure to be within the acceptable limits of the June 15, 1982 revision of Standard Review Plan (SRP) Section 15.3.3.-15.3.4. The results of this analysis further indicate that about 22% of the fuel rods reach a DNBR less than 1.3 and about 15% of the fuel rods reach a DNBR less than 1.0. This occurs for a very short time period (about 2 seconds). Peak clad surface temperature is 1522°F. The licensee indicates that the peak clad surface temperatures are below the threshold for metal-water reaction, and therefore, the results are acceptable.

Reference 6 contains the licensee's analysis of this event at reduced operating pressure. While the resulting peak pressure is lower than in the FSAR analysis, the results of the DNB calculations are more severe, predicting that 63% of the fuel rods reach a DNBR of less than 1.3. The licensee indicates that this analysis was performed on a highly conservative basis, since the coolant pressure increase as a result of the transient was ignored, and rods for which the fluid conditions are beyond the range of the DNB correlation were assigned DNB ratios less than 1.3. In view of the high percentage of potentially damaged fuel as a result of this postulated accident, the staff has performed independent site boundary calculations to determine whether the radiological consequences of the postulated accident meet the guidelines of 10 CFR Part 100. The licensee has implemented standard technical specification (STS) limits for primary coolant iodine. Assuming primary coolant STS limits and 63% fuel cladding damage, the radiological consequences at the site boundary would be less than a small fraction of the 10 CFR Part 100 guidelines. The licensee's analysis did not assume loss of offsite power (LOOP) and thus the radiological consequences could conceivably be higher if LOOP occurred. Therefore, a limiting calculation was also performed assuming that all the fuel cladding is damaged. The resulting site boundary dose is still less than the 10 CFR Part 100 guidelines, and thus meets the acceptance criteria of the June 15, 1982 revision of SRP Section 15.3.3-15.3.4 for site boundary dose.

With regard to the effect of the replacement steam generator, reference 2 indicates that changes in coolant temperature due to secondary parameter changes would not be detected in the core during the time frame of interest for this transient. We concur with this and conclude that the locked rotor analysis is acceptable.

4. Steam Line Break

The FSAR steam line break analysis was performed using 7 combinations of break sizes and initial plant conditions, including large breaks upstream and downstream of the flow limiting nozzle, one and two-loop operation, offsite power available and unavailable, and a break equivalent to steam release through one steam generator safety valve. The analyses were performed assuming end of core life, hot shutdown with the most reactive rod stuck in its fully withdrawn position, and one safety injection pump failing to function. The most severe case involves a break upstream of the flow limiting nozzle, two loops in operation, and loss of offsite power, and results in a peak power after return to criticality of 24%. For the break downstream of the flow measuring nozzle, peak power after return to criticality was of the order of 10%. Utilizing the MacBeth critical heat flux correlation provided acceptable DNBR values for all the transients analyzed.

For operation at reduced pressure and temperature, Reference 6, indicates that, as a result of slightly less stored energy in the coolant system, cooldown is slightly faster and the resulting thermal power is about 1% higher. Minimum DNBR is still above 1.3.

The replacement steam generator lower assembly will provide a slightly higher secondary mass and could thus result in slightly greater cool-down when compared with the original analyses for identical breaks. However, this will be more than offset by the installation of integral flow limiters nozzles in the steam generator cutlet nozzles. Consequently, the FSAR analysis for a break upstream of the flow limiter is bounding for all cases. We conclude that the steam line break analysis are acceptable.

5. LOCA

Reference 2 states that the most applicable existing large-break LOCA analysis to be used for evaluation of the steam generator replacement was performed with 18 percent tube plugging and peaking factor (F_q) equal to 2.32. References 7 and 8 contain such analyses for operating pressures of 2250 psia and 2000 psia, respectively. Reference 4 contains our evaluation of the LOCA analysis submitted in Reference 7. It is concluded that a large-break LOCA when operating Point Beach Unit 1 at a primary pressure of 2250 psia and with up to 18% tube plugging would result in a peak clad temperature (PCT) of 2053^{o}F. Reference 8 provides a LOCA analysis for reduced pressure operation. PCT is predicted at 2062^{o}F. As described in Reference 4, a correction factor of 60^{o}F should be added to these numbers to account for upper plenum injection. The criteria of 10 CFR Part 50.46 are met for both analyses.

Reference 2 also indicates that the replacement steam generators would improve the system performance for the limiting break LOCA compared to a postulated LOCA with the existing steam generators with 18% tube plugging, due to greater primary flow. Also, the replacement steam generators would improve the systems performance for the limiting break LOCA analysis compared to the original unplugged steam generator because of greater primary flow. The tube resistance for the replacement steam generators would be lower than that of the original steam generators at the existing analysis conditions. Therefore, there would be an improvement in the core reflood rates. With regard to the small break LOCA, Reference 1 indicates the worst case small break to be 6" pipe, resulting in a PCT of 1367°F. Reference 2 indicates that the replacement steam generator would have an insignificant effect on the small break LOCA.

We concur with the licensee's conclusion regarding both the small and large break LOCA. We conclude that the licensee's LOCA analysis meets 10 CFR Part 50.46 criteria and is therefore acceptable.

8.1 Radiological Consequences of Postulated Accidents

8.1.1 Accidents During Operation with Repaired Steam Generators

The repaired steam generators will not significantly affect the dose consequences of accidents involving the secondary system. The accidents involving significant dose consequences are the main steam line failure, steam generator tube failure and control rod ejection. The only design change that affects the accident dose consequences is a 2% increase in the volume of the secondary side of the steam generator. The reactor coolant system parameters which affect these accidents will not be changed significantly by the repaired steam generators. These parameters include reactor coolant leakage to the secondary system and the reactor cooldown period. The major dose contribution is from reactor coolant leakage into the secondary system during the accidents.

Specifically, in both the steam generator tube failure and control rod ejection accidents, the increased volume of the secondary system provides for more dilution of the activity which leaks from the reactor coolant side. Because the reactor coolant system parameters have not changed, the total reactor coolant side release time and volume will not change. Therefore, the increased secondary volume should result in a negligible change in doses.

Similarly, the reactor coolant system parameters which affect the main steam line failure accident also remain unchanged. Assuming the same concentration of radionuclides (pre-existing inleakage of reactor coolant), the increased mass of the secondary side will result in a slight increase in offsite doses. The contribution to the doses from additional reactor coolant inleakage during the accident itself would be unchanged. Because the secondary volume increases by 2% and most of the dose is a result of "fresh" reactor coolant inleakage, the total offsite dose will increase by much less than 2%. This slight increase in total offsite dose will not result in estimated consequences in excess of the 10 CFR Part 100 guidelines, and the conclusions concerning these accidents reached in the July 15, 1970 Safety Evaluation for the Point Beach Nuclear Plant Unit No. 1 are not changed due to the repair of the steam generators since the effect of the small secondary volume increase on the evaluation of relevant accident consequences remains very small.

8.1.2 Accidents During the Repair Effort

Rigging Accidents - Impact on Safety-Related Systems/Structures/Components

The licensee has stated that precautions will be taken to preclude the possibility that a rigging or transportation accident will result in damage to any system, structure, or component important to safe operation and maintenance of either Point Beach unit. These precautions include training of equipment-operating personnel, additional protection of buried piping and duct banks where necessary along the steam generator transfer paths, controls on transfer paths and equipment speed, and controls on lift heights, travel directions, location, and swing arcs for both loaded and unloaded cranes. Four structures/components are vulnerable to possible impact during a rigging accident due to the drop of a steam generator lower assembly. They are the containment, Service Building, Extension Building, and overhead electrical wires passing over the work area within range of the crane boom. These items are not required to perform safety-related functions during steam generator repair on that unit. Additionally, the fuel will be in offloaded configuration during the steam generator repair period. The licensee has, therefore, concluded that no evaluation need be performed for damage to these structures/components. We concur with this conclusion and, further, conclude that there will be no radioactive release to the environment from these hypothetical accidents.

Steam Generator Lower Assembly Drop

The steam generator lower assembly, after having been secured, can undergo a hypothetical accidental drop to ground during removal through the equipment hatch, or during transport to the storage building. If such a drop should occur outside containment, the welded plate over the primary side might be breached. To assess the radiological consequences of such an accident, the staff has made a number of conservative assumptions. If it is assumed that ten percent of the solid radioactive corrosion products contained within the steam generator are released following impact (30 curies), and that of this amount one percent will consist of particulates of diameter less than one micron, the resulting maximum radiological consequence at a receptor location on the Lake Michigan shoreline 112 meters from the drop position would be 268 mrem, (assuming a diffusion and transport atmospheric relative concentration of $7x10^{-3}$ sec/m^3). This is a very small fraction of the guideline lung dose limit inferred from ICRP-26 and the 10 CFR Part 20 guideline of 5 rem. A similar drop occurring inside containment would result in a substantially lower dose because of the very circuitous path to the environment.

Cutting of the Reactor Coolant Piping

For this cutting, performed with a welding torch, the staff has conservatively postulated total vaporization of the radioactive corrosion products in the kerf area, some 8.9×10^4 microcuries. Even if it is assumed that all of this radioactivity can be inhaled (without benefit of filtration and aerosol formation and deposition) the lung dose would be 3 mrem, an even smaller fraction of the 10 CFR Part 20 inferred lung dose guidelines.

Accidents Initiated by External Events

No combustibles will be stored in the steam generator storage building. Thus, fire in this building with any subsequent releases of contained radioactive corrosion products is not credible.

The steam generator storage building is located at elevation 29.5 feet above Lake Michigan. The maximum lake surge is, therefore, not expected to be able to initiate a flooding condition in the storage building area. Further there are no streams or other sources of water capable of causing flooding in the building area.

The steam generator storage building will be constructed of reinforced concrete with walls and roof at least two feet thick and a concrete basemat, of construction similar to that of the auxiliary building. It is to be built according to the requirements of the Uniform Building

Code for Zone 1. It is not qualified as Seismic Category I, or designed for tornado/tornado missile resistance. Even if the building were to collapse onto the stored steam generator lower assembly, it is not expected that radioactive particulate crud released to the environment would exceed that for the lower assembly drop accident. The same argument would apply for tornado strike effects and tornado missile impact. Additionally, no stored lower assembly section is likely to become airborne in a tornado due to the massive weight of the assembly.

The staff judges, therefore, that accidents initiated by external events would not result in offsite radiological consequences exceeding those of a steam generator lower assembly drop outside containment, which is well within guidelines inferred from 10 CFR Part 20.

9.0 PHYSICAL SECURITY ASPECTS

Wisconsin Electric Power Company by letter dated November 10, 1982 as revised December 2, 1982 submitted a proposed new Chapter 10 to the Point Beach Nuclear Plant Physical Security Plan in accordance with the provisions of 10 CFR 50.54(p). The staff transmitted their approval of proposed physical security plan change by letter dated December 13, 1982. The details of the plan change are Safeguards Information and are being withheld from public disclosure.

10.0 CONCLUSION

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

Date: **SEP 30 1983**

Attachment 1: Estimate of Personnel Radiation Exposure For Steam Generator Replacement Operations at Point Beach Unit 1.

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REFERENCES

1. Point Beach Nuclear Plant Units 1 and 2 Final Safety Analysis Report (updated).

2. Point Beach Nuclear Plant Unit 1 "Steam Generator Repair Report" August 1982.

3. NRC Confirmatory Order for Modification of Point Beach 1 License, November 30, 1979.

4. NRC Modifying Confirmatory Order of November 30, 1979, dated January 3, 1980.

5. NRC Letter WE dated April 29, 1980, forwarding Amendments No. 44 and 49 to Point Beach Unit 1 Facility Operating License.

6. WCAP 8151, Point Beach Unit 2 Low Pressure Analysis, June 1973.

7. WE Letter of November 19, 1979, forwarding ECCS Reanalysis for 18% Steam Generator Tube Plugging Limit, Point Beach Unit 1.

8. WE Letter of November 27, 1979, forwarding Low Pressure ECCS Evaluation for 18% Steam Generator Tube Plugging, Point Beach Units 1 and 2.

9. WE Letter of November 9, 1982, forwarding responses to our request for additional information regarding the LOCA and locked rotor analyses.

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR STEAM GENERATOR REPLACEMENT OPERATIONS AT POINT BEACH UNIT - I

PHASE-I SHUTDOWN AND PREPARATORY ACTIVITIES

TASK DESCRIPTION		ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
I	Shutdown and Preparatory Activities	58,887	237.3
II	Removal Activities	141,680	421.7
III	Installation Activities	334,138	605.8
IV	Post Installation and Startup Activities	87,700	118.3
V	Steam Generator Storage Activities	1,532	6.6
	PROJECT TOTALS	623,937 (All Tasks)	1389.7

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR STEAM GENERATOR REPLACEMENT OPERATIONS AT POINT BEACH UNIT - I

PHASE-I SHUTDOWN AND PREPARATORY ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Install Polar Crane Gibpole Modification	2,000	10.0
Installation of Jib Cranes	5,584	6.7
Misc. Disassemble Manipulator Crane	1,000	2.0
Install Steam Generator Transport System	5,738	6.7
Removal Constainment Obstructions	2,000	3.5
Installation of Temporary Ventilation System	2,000	2.5
Temporary Scaffolding	5,000	29.7
Temporary Lighting and Power	2,000	2.0
Cleanup and Decon	10,712	35.0
Polar Crane Operator	1,000	2.0
Shielding	11,500	100.0

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR STEAM GENERATOR REPLACEMENT OPERATIONS AT POINT BEACH UNIT-I

PHASE-I: SHUTDOWN AND PREPARATORY ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
H.P., Q.A	12,723	15.0
Miscellaneous	2,000	5.0
Installation of Service Air System	630	2.2
Work Platform Modification	2,000	1.0
Protection of Containment Components	1,500	8.0
Project Supervision and Administration	1,000	6.0
SUBTOTAL PHASE I	58,887	237.3

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR
STEAM GENERATOR REPLACEMENT OPERATIONS
AT POINT BEACH UNIT-1
PHASE II: REMOVAL ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Removal of Insulation (lower shell, RC piping)	1,224	9.3
Removal of Insulation (upper shell, mainstream and feedwater piping)	446	3.2
Removal of Miscellaneous Piping	3,356	24.4
Set Up Steam Generator Girth Cut Equipment	600	2.0
Cut and Remove Steam Generator Upper Shell	3,536	6.8
Cutting of Reactor Coolant Piping	9,139	96.9
Cutting of Mainstream and Feedwater Piping	1,412	2.4
Disassembly of Steam Generator Supports	5,910	34.7
Removal of Moisture Separation Equipment	3,794	8.1
Refurbish Steam Generator Upper Shell	11,543	10.2
Removal of Steam Generator Level Instruments and Blowdown Piping	1,892	4.7
Removal of Steam Generator Lower Shell	2,733	17.6

ESTIMATE OF PERSONNEL RADIATION
EXPOSURES FOR STEAM GENERATOR
REPLACEMENT OPERATIONS AT
POINT BEACH UNIT-1
PHASE II: REMOVAL ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Temporary Scaffolding	8,227	29.1
Temporary Lighting and Power	3,810	5.2
Cleanup and Decon	35,731	86.2
Polar Crane Operator	1,837	1.7
H.P., Q.A.	20,167	42.4
Material Handling, Equipment Maintenance, and Miscellaneous Construction Activities	16,323	3.9
Project Supervision and Administration	10,000	12.9
SUBTOTAL PHASE II	141,680	421.7

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR STEAM GENERATOR REPLACEMENT OPERATIONS AT POINT BEACH UNIT-I

PHASE III: INSTALLATION ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Steam Generator Lower Shell Installation	7,744	12.7
Installation of Reactor Coolant Piping	43,666	193.1
Steam Generator Girth Weld	19,135	9.9
Installation of Main Steam Piping	6,661	6.9
Installation of Feedwater Piping	4,715	2.3
Installation of Blowdown and Miscellaneous Piping	12,252	45.2
Install Steam Generator Level Instruments	7,622	9.6
Installation of Insulation	7,747	25.0
Temporary Scaffolding	12,148	35.6
Temporary Lighting & Power	6,802	6.5
Cleanup and Decon	79,563	127.7
Polar Crane Operator	5,245	2.1
H.P., Q.A.	73,061	66.2
Material Handling, Equipment Maint., and Misc. Construction Activities	35,777	25.5
Project Supervision & Administration	12,000	37.5
SUBTOTAL PHASE III	334,138	605.8

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR
STEAM GENERATOR REPLACEMENT OPERATIONS AT
POINT BEACH UNIT-I
PHASE-IV: POST INSTALLATION AND STARTUP ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Install Biological Shield Wall	2,121	2.9
Repair Crane Wall Opening	214	0.4
Install S/G Recirculation & Transfer System	16,534	33.5
Remove Polar Crane Gibpole Mod.	1,500	5.0
Install Reactor Cavity Coaming	650	0.7
Reassemble Manipulator Crane	1,256	1.4
Remove S/G Transport System	200	1.5
Hydrostatic Tests	2,376	3.4
Temporary Scaffolding	3,382	6.4
Temporary Lighting & Power	1,712	1.5
Cleanup and Decon	14,378	23.2
Polar Crane Operator	1,186	0.5
Painting	9,000	8.0
H.P., Q.A.	14,321	9.8
Miscellaneous	3,000	5.0
Material Handling, Equipment Maint., and Miscellaneous Const. Activities	10,000	9.7
Project Supervision & Administration	5,870	5.4
SUBTOTAL PHASE IV	87,700	118.3

ESTIMATE OF PERSONNEL RADIATION EXPOSURES FOR STEAM GENERATOR REPLACEMENT OPERATIONS AT POINT BEACH UNIT-1

PHASE-V: STEAM GENERATOR STORAGE ACTIVITIES

TASK DESCRIPTION	ESTIMATED LABOR (PERSON-HOURS)	ESTIMATED EXPOSURE (PERSON-REM)
Steam Generator Storage Activities	1,532	6.6

UNITED STATES NUCLEAR REGULATORY COMMISSION

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-266

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 75 to Facility Operating License No. DPR-24, issued to Wisconsin Electric Power Company (the licensee), which revised the license for operation of the Point Beach Nuclear Plant Unit No. 1 (the facility) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendment was effective as of the date of its issuance.

The amendment allows repair of steam generators by replacement of major components.

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 rule 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 July 12, 1982 (47 FR 30125). A request for hearing was filed on August 10, 1982 by Wisconsin's Enviromental Decade. Following a Special Prehearing Conference held on November 19, 1982 on this matter, the Atomic Safety and Licensing Board (ASLB) dismissed WED's Petition for Leave to

Intervene by Order dated December 10, 1982. The ASLB cited WED's failure to proffer one good contention with adequate bases and WED's willful absence from the Special Prehearing Conference as grounds for dismissal. Though appealed by WED, the ASLB Order has been upheld by the Atomic Safety and Licensing Appeal Board (ASLAB) in their March 22, 1983 Decision.

The Commission has prepared an Environmental Impact Statement related to the action and has concluded that there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the Facility dated May 1972.

For further details with respect to the action see (1) the application for amendment dated May 27, 1982, as supplemented July 27, August 9, 1982 and March 1, 1983, (2) Amendment No. 75 to License No. DPR-24, (3) the Commission's related Safety Evaluation, and (4) Final Environmental Impact Statement. 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 Joseph P. Mann Public Library, 1516 16th Street, Two Rivers, Wisconsin. 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 Licensing; a copy of item (4) may be purchased at current rates from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161, and from the Sales Office, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

Dated at Bethesda, Maryland this 30th day of September, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing