NOV 1 0 1981

Docket No. 50-266

RECEIVED
NOV 18 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. Sol Burstein
Executive Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Burstein:

The Commission has issued the enclosed Amendment No. 56 to Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated July 2, 1981 as modified by letter dated October 12, 1981, and which was the subject of the Atomic Safety and Licensing Board's Memorandum and Order dated November 5, 1981.

This amendment authorizes power operation with up to six tubes in one steam generator having degradation exceeding the plugging limit provided these tubes have been repaired by insertion of sleeves to bridge the degraded or defective portion of the tube. It also establishes an additional plugging limit for these six repaired tubes of 35% degradation of the sleeve wall nominal thickness. This limit has been discussed with members of your staff and was found acceptable.

Section 2.2 of the Safety Evaluation Report (SER) states that some additional processing of finite element stress data must yet be performed before the structural and fatigue analyses can be evaluated against the 3 Sm limit for primary plus secondary stress. The NRC staff received this information by letter dated November 6, 1981 and our review confirms our preliminary findings that Code allowables have been met. This discussion will be included in our Safety Evaluation of the full scale sleeving program.

8112030587 811110
PDR ADOCK 05000266
P PDR

DEC 1 0 1981

OFFICE							
SURNAME							
DATE							

Copies of the Safety Evaluation, our related Environmental Impact Appraisal and the combined Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Original signed by:

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 56 to DPR-24
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice of Issuance and Negative Declaration

cc: w/enclosures
See next page

Distribution

Docket Files (2)	CMiles
NRC PDR (2)	RDiggs
Local PDR	TColburn
TERA	ACRS (10)
NSIC	RAClark
ORB#3 Reading	PKreutzer (3)
DEisenhut	BScharf (10)
OELD	RBallard
I&E (4)	Chairman, ASLAB
GDeegan (8)	Gray Files (+4)
DBrinkman	

* See previous page for concurrence.

OFFICE	ORB#3:DL*	ORB#3:DL*	ORB#3:DL*	AD:OR:DL*	OELD*		
SURNAME	PMKreutzer	TColburn/pn	RAClark	TMNovak	RBachmann		
DATE	11/6/81	11/6/81	11/6/81	11/9/81	11/10/81		

against WED's motion for continuance and ruled that WED had not provided sufficient material basis of fact to show cause why the amendment should not be issued. Further, the licensee would be allowed to operate at power with up to six steam generator tubes which have degradation exceeding the plugging limit, provided that the tubes have been repaired by sleeving subsequent to an acceptable finding by the NRC staff.

Copies of the Safety Evaluation, our related Environmental Impact Appraisal and the combined Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No.   to DPR-24
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice of Issuance and Negative
   Declaration

cc: w/enclosures
    See next page

Distribution

Docket Files (2)	CMiles
NRC PDR (2)	RDiggs
Local PDR	TColburn
TERA	ACRS (10)
NSIC	RAClark
ORB#3 Reading	PKreutzer (3)
DEisenhut	BScharf (10)
OELD	RBallard
I&E (4)	Chairman, ASLAB
GDeegan (8)	Gray Files (+4)
DBrinkman	

Subject to changes noted.

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD		
SURNAME	PMKreutzer	TColburn/pn	RAClark	TMNovak	Bachmann		
DATE	11/6/81	11/6/81	11/6/81	11/9/81	11/10/81		

Wisconsin Electric Power Company

cc:
Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Joseph Mann Library
1516 Sixteenth Street
Two Rivers, Wisconsin 54241

Mr. Glenn A. Reed, Manager
Nuclear Operations
Wisconsin Electric Power Company
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Ms. Kathleen M. Falk
General Counsel
Wisconsin's Environmental Decade
114 N. Carroll Street
Madison, Wisconsin 53703

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 S. Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming
dtd: 7/2/81, 10/12/81

Chairman
Public Service Commission of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Mr. William Guldemond
USNRC Resident Inspectors Office
6612 Nuclear Road
Two Rivers, Wisconsin 54241

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. 56
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 July 2, 1981 as modified by letter dated October 12, 1981, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 10, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NOS. 50-266

Revise Appendix A as follows:

Remove Page	Insert Page
15.4.2-1c	15.4.2-1c

Defect is an imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which the tube must be removed from service, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

B. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged prior to return to power from a refueling or inservice inspection condition.*

C. Reports

1. After each inservice examination, the number of tubes plugged in each steam generator shall be reported to the Commission as soon as practicable.

2. The complete results of the steam generator tube inservice inspection shall be included in the Operating Report for the period in which the inspection was completed. In addition, all results in Category C-3 of Table 15.4.2-1 shall be reported to the Commission prior to resumption of plant operation.

3. Reports shall include:

   (a) Number and extent of tubes inspected

   (b) Location and percent of all thickness penetration for each indication

   (c) Identification of tubes plugged

4. Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection - shall provide the information required by Specification 15.4.2.C.2 and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

B. In-Service Inspection of Reactor Coolant System Components Other Than Steam Generator Tubes

The in-service inspection program is generally based on the recommendations of ASME Boiler and Pressure Vessel Code, Section XI, Summer 1971 Addenda, as practical for a plant whose design and construction preceded issuance of the recommendations. The commitments herein are made assuming that the necessary inspection

---

*Point Beach Nuclear Plant Unit 1 may be operated at power with up to six tubes in one steam generator having degradation exceeding the plugging limit provided those tubes have been repaired by insertion of sleeves into the tubes to bridge the degraded or defective portion of the tube. The plugging limit is 35% of the nominal sleeve wall thickness for tubes that have been repaired by sleeving.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 56 TO FACILITY OPERATING LICENSE NO. DPR-24

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-266

TABLE OF CONTENTS

1.0 INTRODUCTION

By letter dated July 2, 1981, Wisconsin Electric Power Company (licensee) submitted an application for license amendments consisting of proposed changes to the Technical Specifications for Point Beach Nuclear Plant Units 1 and 2. These proposed Technical Specification changes would allow operation at power of Units 1 and 2 with steam generator tubes having degradation exceeding the plugging limit (40% nominal wall thickness) provided these tubes have been repaired by insertion of sleeves into the tubes to bridge the degraded or defective portion of the tubes. The proposed issuance of these amendments was prenoticed in the Federal Register on August 7, 1981 due to the strong public interest on this subject.

The licensee also submitted by letter dated October 12, 1981, a modification to their proposed license amendment for Unit 1 dated July 2, 1981. This modification proposed Technical Specification changes to allow operation of Unit 1 at power with up to six tubes in one steam generator having degradation exceeding the plugging limit provided these tubes have been repaired by insertion of sleeves into the tubes to bridge the degradated or defective portions of the tubes. The licensee also plans to sleeve six tubes having degradation less than the plugging limit. The licensee's stated reason for submitting this modification is to conduct a demonstration sleeving program on Point Beach Unit 1 during the October 9, 1981 refueling outage. This demonstration program will utilize two separate sleeving processes and the licensee hopes it will provide valuable information and experience for use during their full-scale sleeving program.

This Safety Evaluation documents the results of the NRC staff's review and evaluation of the licensee's proposed demonstration steam generator tube sleeving program including the environmental and radiation exposure impact.

2.0 DISCUSSION

2.1 Sleeving Process Description

The sleeving demonstration program scheduled for the fall 1981 refueling outage of Point Beach Unit 1 is expected to include removal of explosive and mechanical plugs from previously plugged tubes where degradation had exceeded the plugging limit in the Technical Specifications. All tubes from which plugs have been removed will be inspected with eddy current techniques throughout their length prior to sleeving. Should indications of progression of degradation, or new indications of degradation be seen outside the proposed sleeved region of the tube, the tube will not be sleeved, but will be plugged in accordance with the Technical Specification requirements.

To provide a technical basis for the proposed sleeving demonstration program, the licensee has submitted Westinghouse Report WCAP-9960 (Proprietary), dated September 28, 1981, and entitled, "Point Beach Steam Generator Sleeving Report for Wisconsin Electric Power Company." The licensee has submitted additional information by letters dated October 9, 16, 24 and 26 in response to questions by the ASLB and the NRC staff. They have also responded to other questions during conference calls with the NRC staff.

The sleeving process consists of installing, inside the steam generator tube, a smaller diameter tube (sleeve) to span the degraded area of the parent tube. The sleeves are intended to restore the integrity of the degraded tubes by providing a new primary pressure boundary which has been sized to the ASME Boiler and Pressure Vessel Code, Section III.

The sleeves are fabricated from thermally treated Inconel 600 tubing to provide a maximum resistance to stress corrosion cracking. The sleeves will be inserted inside the existing tube (mill annealed Inconel 600) and joined to the tube ID at the upper and lower sleeve ends. The sleeves will span the distance from the tube inlet to a few inches above the top of the tubesheet. The Point Beach sleeves are intended to address the general intergranular attack and stress corrosion cracking which has been confined to the tubesheet area.

The sleeves used in the demonstration program will employ two different upper sleeve joint designs. The "reference" upper joint design is a structural joint which provides a leak limiting seal. A functional requirement for "reference" upper joints is that they must be sufficiently leak limiting such that the total leakage between the primary and secondary for all the sleeves taken together is less than the Technical Specification leak rate limit during normal operation. In addition, total leakage must be maintained to within tolerable limits during postulated accidents. The acceptance criteria imposed during verification leak testing of the joint is based upon these total leakage limits divided by the total number of tubes eventually planned for sleeving (approximately 2500 tubes).

The second or "alternate" upper joint design is also a structural joint. This joint is fabricated using a proprietary heating process to form a leak tight seal. The lower sleeve joint also provides a structural and leak tight seal, but is not fabricated with the proprietary heating process.

The Point Beach sleeves and sleeve joints are basically similar to those at San Onofre Unit 1 from the standpoint of design and joint fabrication techniques. The San Onofre sleeves have been extensively tested for structural, metallurgical, corrosion, and leak tight (or leak limiting) integrity.

2.2 Structural Verification Analyses

Structural analyses of the sleeved tube assembly are being performed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. These analyses are intended to demonstrate adequate fatigue performance and structural margins for the full range of normal operating, transients, and accident (e.g., LOCA, MSLB) condition loadings. The structural and fatigue analyses include consideration of stresses in the sleeved tube assemblies which could result from hourglassing (deformation) of the support plate flow slots, and from flow induced vibration. The analyses have essentially been completed; however, some additional processing of finite element stress data must yet be performed before they can be evaluated against the 3 Sm limit for primary plus secondary stress. The preliminary results submitted by letter dated October 24, 1981, indicate the Code allowables for primary membrane, primary membrane plus bending stress, and fatique usage have been met.

Strength analyses have been performed to establish the minimum wall thickness requirement (or allowable wall degradation) to assure compliance with the Regulatory Guide 1.121 "no yield" criterion under normal operating conditions. These analyses have also established the minimum wall thickness requirements (and allowable wall degradation) to preclude a gross tube burst under the pressure loadings associated with a postulated MSLB accident, consistent with the Regulatory Guide criterion and the Code limits on primary membrane stress under faulted conditions. The results of these analyses will be used to set the Technical Specification plugging limit for the sleeves.

2.3 Verification Testing of Sleeve Joints

The structural analyses of the sleeved tube assemblies are being supplemented by extensive mechanical testing to verify acceptable structural strengths, fatigue performance and leaktight integrity of the upper and lower joints. The test mockups for the lower joint include tubesheet mockups from which the effects of removing both mechanical and explosive type plugs have been simulated. The joints have been formed using the same fabrication techniques and parameters as will be used in the field. Each of the joints is being subjected to axial load (to simulate loads caused by differential thermal expansion) and pressure cycling tests to verify the long term sealing integrity of the joints under the specified operating transients (e.g., heatup/cooldown and plant loading/unloading cycles). Specimens for each type joint will also be tested to the maximum pressure and axial load levels expected during postulated accident conditions. For each of the three joint designs, testing has proceeded to as much as the equivalent of five years of operation with no adverse findings reported to date. Further testing is in progress and will be continued for an equivalent 35 years of operation.

Similar mechanical tests have been completed for the San Onofre joints to support thirty years operation with the results indicating acceptable structural and leak limiting performance.

2.4 Verification of "Leak Before Break"

Westinghouse tests indicate that margin to burst exists at the MSLB pressure differential for a through wall crack which is leaking at less than the Technical Specification limit during normal operation. The tests indicate that the required through wall crack length for a tube burst under MSLB conditions is .5 inches, whereas a through wall crack longer than .4 inches will result in leakage in excess of the Technical Specification leakage rate limit during normal operation.

2.5 Effect of Proprietary Heating Process on Upper Alternate Joint Integrity

The proprietary heating process for the "alternate" upper joint design will result in some degradation of the mechanical properties of the sleeve and tube wall material local to the seal between the sleeve and the tube.

Tensile tests of individual San Onofre tube and sleeve specimens following a simulated joint heating process indicated a significant reduction in the ultimate and yield strength at the location where the peak temperature had been reached. This corresponds to the center of the region where the tube and sleeve would be sealed. As evidenced by variations in hardness and grain size measurements as one proceeds away from this location, heat process effect on the yield and ultimate strength is localized to within the width of the seal. Tensile tests of a number of joint specimens resulted in tensile failures of the sleeve wall invariably between two and three inches below the sealed location, at levels in excess of minimum requirements (Ref. 1). Westinghouse has also reported that the stress strain curve of the "alternate" upper joint almost duplicates that of virgin Inconel 600 material.

Westinghouse has reported that confirmatory tests for the actual Point Beach "alternate" joint configuration have indicated similar results and that the overall joint strength exceeds Code requirements.

Internal pressure tests to three times normal operating pressure, and external pressure tests to 1.5 times the maximum LOCA pressure loading resulted in no failures for the San Onofre "alternate" upper joint specimens. Similarly, load cycling tests (to simulate pressure plus thermal cycling) for the expected number of operating cycles over a 30 year lifetime were completed with no failures. Similar confirmatory tests are in progress for the actual Point Beach configuration, with the exception of the collapse test.

2.6 Discussion of Corrosion Aspect and Verification Testing

The corrosion that has occurred on the outer surface of the tubes has been attributed to caustic corrosion resulting from the use of phosphate water chemistry in the secondary water with massive phosphate additions and the formation of caustics due to impurities from persistent leaky tubes in the steam condenser. The chemistry control program of the secondary side water was switched to an all-volatile treatment in September of 1974, though free hydroxide continued to be present in the blowdown water until 1978.

Most of the steam generator tube corrosion and degradation has occurred in the central region of the inlet end of the tube bundle. Some intergranular stress corrosion cracking, wastage, and thinning has occurred at a location just above the tubesheet in the sludge zone, but the more extensive intergranular corrosion has occurred in the tubesheet crevices. Although the licensee's tube degradation rate has slowed recently, tube degradation could continue.

We have reviewed the corrosion test program performed in support of the Southern California Edison (SCE) plant, San Onofre Unit 1. This work was cited by the licensee in support of the present application request. The corrosion tests performed were extensive, involving the use of capsule tests and modified boiler tests in which the environment that existed in San Onofre Unit 1 was simulated and its effect on the sleeved tubes was studied. The environment in the tubesheet crevice at Point Beach Unit 1 is similar. An extensive test program was performed studying the effects of caustic on the corrosion resistance and stress corrosion cracking of the sleeving material. Confirmatory testing of the corrosion and stress-corrosion cracking resistance of both the upper and lower joints of the Point Beach configuration is in progress.

2.7 Eddy Current Test Capabilities

Eddy current data is provided in the Repair Report to demonstrate the applicability of the conventional bobbin type ECT probe to the inspection of the sleeved tube assemblies. (This data was actually obtained for San Onofre sleeved assemblies.) At the optimum test frequency for the sleeve, the amplitudes of the ECT signals ranged from 70% to 100% of those for a non-sleeved tube for calibration holes of 40% and 100% throughwall depth, respectively. This data is indicative of the relative flaw sensitivity outside the tubesheet, whereas most of the sleeve length will be located within the thickness of the tubesheet. The Westinghouse investigation indicates that within the thickness of the tubesheet the "signal to noise ratio" associated with a sleeving defect is substantially more than that associated with a flaw in a non-sleeved tube. Thus, Westinghouse has concluded that the sleeve in the tubesheet region will have a higher degree of inspectability than an unsleeved tube in this region.

The inspectability of the tube wall is of interest at and above the upper sleeve joints. The Westinghouse study indicates that the amplitude of the ECT signals for calibration holes in excess of 40% through wall were approximately 50% of those for non-sleeved tubes at a test frequency of 100 KHZ. At a test frequency of 350 KHZ, the amplitude sensitivity was reduced to approximately 30% to 40% of that for a non-sleeved tube.

Eddy current inspection of the sleeve joints will present some difficulties particularly for the "alternate" type upper joint. The sleeve joints contain a number of features which will produce competing ECT signals making it more difficult to discriminate sleeve or tube wall defects at these locations. The application of the multifrequency techniques will provide enhanced capability to discriminiate flaw signals from these competing signals. Westinghouse is currently investigating ECT procedures to further improve the inspectability of these regions including the use of magnetic bias techniques and alternate probe types such as the crosswound probe, the rotating pancake (RPC) probe, and the multicoil surface riding probe.

3.0 EVALUATION

3.1 Structural and Leak Tight Integrity

We have reviewed the extensive program of verification analysis and tests to qualify the structural and leak tight (or leak limiting) integrity of the sleeved tube assemblies and the results thus far available. Although an assessment of primary plus secondary stresses against the 3 Sm limit ("shakedown") of the ASME Code remains to be completed, the licensee has sufficiently demonstrated by analysis that adequate margin will exist against a burst failure of the sleeve during the full range of normal, transient, and postulated accident conditions, consistent with the primary membrane and primary plus bending stress limits of the Code. Mechanical load cycling tests to verify the long term structural, fatigue, and leak tight (or leak limiting) performance of the sleeve joints have reached the equivalent of five years of operation with no adverse results. This preliminary data, coupled with the results of the fatigue analysis performed to the ASME Code requirements, provides reasonable assurance against a fatigue or shakedown failure of the demonstration sleeve joints during the interim period before the remaining analytical effort and testing is complete.

Regarding this sealing integrity of the joints, even if the demonstration sleeve joints should leak (between the sleeve and tube wall) at several orders of magnitude higher than what has been indicated by the test results thus far, the total leakage would be insignificant compared to the licensee's criteria for allowable total leakage. This is due to the relatively small number of sleeves involved in the demonstration program and the inherent leak limiting geometry of the sleeve joint.

We have also reviewed the licensee's "leak before break" analysis. We find that the available margins are consistent with those which exist for the original tubing and are acceptable.

3.2 Plugging Limit

The licensee has not yet proposed a plugging limit for the sleeves should they become degraded. Based upon our review and assessment of the minimum wall thickness requirements calculated by Westinghouse, we find that a 35% plugging limit (sleeves with greater than 35% through wall degradation due to be plugged) will assure acceptable margins to failure consistent with the criteria of Regulatory Guide 1.121. Pending additional information from the licensee to justify a less restrictive limit, we are imposing a 35% plugging limit as an interim requirement.

3.3 Alternate Upper Joint Integrity

Laboratory testing has shown a significant reduction in the ultimate and yield strength of the sleeve and tube material in the zone local to where the sleeve wall is sealed to the tube wall. However, tensile tests of the San Onofre and Point Beach joint configurations have demonstrated that the sleeve and tube wall at the seal will reinforce each other and that the overall strength of the joint exceeds that of a sleeve wall exhibiting a tensile strength equal to the design minimum strength in the ASME Code. Based upon this, the extensive mechanical tests (proof pressure tests, pressure and axial load cycling tests) which have been completed for San Onofre, and the confirmatory testing which has been completed to date for the actual Point Beach joint configuration, we conclude that there is reasonable assurance against a structural failure of the joint during the interim period before all tests are completed. Primary side and secondary side hydrotests will be performed on the sleeved tube assemblies subsequent to the sleeving operation and provide additional assurance of joint integrity.

We have also reviewed the difficulties experienced at San Onofre regarding localized erosion of the sleeve and tube wall at the joint as a result of the heating process. Based upon the metallographic examinations which have been performed on the San Onofre joints and revised heating parameters which have been implemented at Point Beach, we have concluded that this phenomenon will not have any significant adverse affect on the integrity of the Point Beach joints. Additional assurance is provided by the on-going mechanical testing of these joints which have been fabricated to the process parameters to be used in the field and the eddy current and hydrostatic tests that will be performed following the sleeving operation.

3.4 Corrosion Resistance

We have reviewed the test data from the San Onofre corrosion program for the sleeve repair and find that the tests and their results are directly applicable to the Point Beach sleeving repair test program. The small difference is the tube dimensions that cause slightly different operating values in the fabrication procedure do not affect significantly the corrosion resistance of the tubes or the joints. The test program has studied the behavior of the repair program materials in pure water, in primary coolant, and in 10% caustic solutions to simulate the continued hide out of caustic in the crevices and sludge on the secondary side of the steam generator. This work has shown that the thermal treatment to be given to the Inconel sleeves is effective in reducing the probability of caustic stress corrosion developing on these sleeves. It has also been shown that the small, controlled amount of cold work performed on the Inconel in attaching the sleeve to the steam generator tube was not sufficient to cause a significant increase in the susceptibility of the tube to stress corrosion cracking from the primary side water. This amount of cold work is significantly less than that which occurred where the tube was expanded into the lower portion of the tubesheet during the original fabrication. To date no cracking has developed in that area in Point Beach, San Onofre, or in model boilers and heat crevice tests. Further the tests have shown that there is only minor degradation of the material properties and corrosion resistance of the tubes at the upper joints. This has been shown by hardness test traverses and corrosion tests in caustic.

3.5 Eddy Current Inspectability

The eddy current inspectability of the sleeve walls between upper and lower joints will be comparable to that for an unsleeved tube without a significant loss of sensitivity. Geometric discontinuities at the sleeve joints will produce signal interference. However, the use of non-standard eddy current probe types and multifrequency techniques should permit adequate inspections of these areas. One local area that may present special difficulties is the sleeve joint which has received the proprietary heating process. Westinghouse is investigating methods to improve the inspectability of this area.

In the meantime, the preservice eddy current inspection of the sleeves will be supplemented by primary side and secondary side hydrostatic tests (2000 psid and 800 psid, respectively) to provide added assurance of the joint integrity.

4.0 ALARA Considerations

The licensee has taken into account ALARA considerations for each of the radiation activities involved in the proposed steam generator sleeving demonstration at Point Beach. ALARA activities specifically directed to reduction of occupational radiation exposures include: decontamination of steam generators, personnel training in full-size mockups, installation of shielding, if necessary, to reduce radiation exposures to repair personnel.

Administrative control of personnel exposures will be effected by careful planning of maintenance procedures for the job, in order to minimize the number of personnel used to perform the various tasks involving relatively high doses and dose rates. TV surveillance of personnel during tasks will be used to identify areas resulting in high exposures, and thus to initiate suitable dose-reducing actions.

Based on prior inplant experience with channel head decontamination and laboratory decontamination, no significant increase in airborne radioactivity is to be expected. However, vapors from the channel head will be drawn through a high efficiency air particulate filtration system before release to the plant filter system. All sleeving operations will be monitored to keep airborne releases to a minimum. The licensee does not expect that auxiliary ventilation or special enclosures will be necessary.

The licensee had made use of experience gained in prior channel head decontamination in planning for the proposed tube sleeving activities. Data was available for Point Beach Unit 1, Takahama Unit 1, San Onofre Unit 1, and Turkey Point Unit 3. In particular, the applicant considered information on mechanisms used in prior decontamination. The licensee has provided information relevant to projected occupational radiation exposures resulting from the demonstration decontamination/sleeving program at Point Beach Unit 1, as well as from the proposed full-scale sleeving program for both units.

The licensee has estimated the radiation doses likely to be associated with the processes involved in the sleeving program:

(a) installation of remoting tools and equipment - 5 person-rems,

(b) decontamination of the steam generator - 10 person-rems (including tube decontamination),

(c) installation of additional shielding, if necessary - 10.6 person-rems (9.5 for the channel head, 1.1 for nozzle shield removal),

(d) inspection and testing - 2.9 person-rems (92 millirems/sleeve eddy current inspection, 300 millirems/sleeve test),

(e) de-plugging tubes for sleeving - 3.4 person-rems/tube (explosive), 0.9 person-rems/tube (mechanical).

(f) sleeving - 4-5 person-rems/tube.

The licensee has provided realistic estimates of dose rates and occupancy factors, as the bases for these dose estimates, and has estimated the total person-rem dose resulting from the demonstration sleeving program at Point Beach Unit 1 at 48-60 person-rems assuming a decontamination factor of about 2.5.

The radiation exposure data and the operational experience resulting from the proposed demonstration of the sleeving process at Point Beach Unit 1 will be a test of proposed radiation control techniques, and will provide a basis for a more refined and more precise estimation of doses likely to result from the proposed future sleeving process of both units.

5.0 REDUCED FLOW CONSIDERATIONS

The licensee has stated that the sleeving of 20 steam generator tubes is equivalent to the reduction in flow through the steam generator caused by plugging one steam generator tube. The licensee plans to sleeve 12 steam generator tubes. Acccording to the licensee's estimates this will cause less effect than plugging one tube.

Further, some of the tubes the licensee plans to sleeve will be tubes previously degraded beyond the plugging limit. The licensee plans to remove the plugs from these tubes and insert sleeves to bridge the degraded or defective portions of these tubes. Based on the licensee's estimates, this would result in a net increase in flow through the steam generators.

Even if the licensee's estimates on the amount of flow reduction associated with sleeving a steam generator tube are in error, and even if the licensee does not recover any previously plugged tubes by sleeving, this will not present an unreviewed safety question for the demonstration sleeving program. Point Beach Unit 1 is operating with an 18% plugging limit for its steam generators. This is based upon an 18% tubes plugged ECCS (Emergency Core Coolant System) analysis submitted by the licensee and approved by the NRC staff. Currently between 12-13% of the steam generator tubes in Unit 1 are plugged. Since 1% of the total number of tubes is approximately 32 tubes for each steam generator, even assuming that the reduction of flow caused by sleeving a steam generator tube was equivalent to that caused by plugging a tube, this is still well within the limits of the previously approved analysis.

For the reasons stated above, the staff finds the effect of the steam generator demonstration sleeving program to be insignificant from a flow reduction standpoint.

6.0 CONCLUSIONS

Based upon the above evaluation, we conclude that the verification analyses and tests completed to date for the Point Beach sleeves, plus the similar program which has been completed for the San Onofre sleeves, provides reasonable assurance that the sleeves and sleeve joints will exhibit acceptable mechanical strength corrosion resistance and leak tight (or leak limiting) capability for the interim period before the Point Beach sleeve verification program is completed. Even if the demonstration sleeves' joints develop substantially more leakage than indicated by test, the total leakage will be insignificant.

The preservice eddy current inspection and primary side and secondary side hydrostatic tests to be performed prior to startup, and the stringent primary to secondary leak rate limits in the Plant License, will provide additional assurance that the sleeved assemblies will maintain adequate tube integrity during normal operation and postulated accidents. If leakage in excess of the leakage rate limit does occur, the plant will be shut down for evaluation of the cause of the leak and appropriate corrective action. Until such time as the licensee submits justification for a less restrictive plugging limit, we require that sleeved tube assemblies containing sleeve indications equal to or greater than 35% through-wall be plugged.

Based on the staff's review of the Point Beach Steam Generator Tube Sleeving Report, and the additional information provided, we conclude that the licensee's estimated dose for this project appears reasonable and that the licensee intends to implement reasonable radiation protection actions that should maintain inplant radiation exposures within the applicable limits of 10 CFR Part 20, and should maintain exposures ALARA.

Based upon the staff's review of the reduced flow considerations associated with the demonstration sleeving project, the staff finds the effects to be within the range of the previously approved ECCS analysis for operation with up to 18% of Unit 1's steam generator tubes plugged. Therefore, the staff finds its impact upon the health and safety of the public to be insignificant.

We have further 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 the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCE:

1. Transcript of "Steam Generator Sleeving Review Board Meeting, San Onofre Unit 1, Steam Generator Sleeve Repair for Southern California Edison, Westinghouse Electric Corporation, Forest Hills Division, Pittsburgh, Pennsylvania, 15221, Thursday, October 23, 1980 - 8:15 A.M., Friday, October 24, 1980 - 8:05 A.M.".

Date: November 10, 1981

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 56 TO OPERATING LICENSE NO. DPR-24

WISCONSIN ELECTRIC POWER COMPANY

DEMONSTRATION PROGRAM OF STEAM GENERATOR

REPAIR BY MEANS OF SLEEVING

POINT BEACH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-266

DATE: October 26, 1981

1.0 INTRODUCTION

Wisconsin Electric Power Company (WE) by letter application dated July 2, 1981, as modified by letter dated October 12, 1981 seeks a license amendment which would authorize WE to operate with six steam generator tubes sleeved rather than plugged which have degradation exceeding the plugging limit defined by Technical Specification 15.4.2.A.5(a) at Point Beach Nuclear Plant Unit 1. This Environmental Impact Appraisal documents the results of the staff review and evaluation of the environmental and radiation exposure impact of the steam generator tube sleeving - demonstration project and interim operation of Unit 1 at power with 12 tubes sleeved (up to six of which have degradation exceeding the plugging limit) until final review of their overall steam generator tube sleeving program has been completed. Based on its review, the Staff finds that the proposed action will not significantly affect the quality of the human environment.

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 Commission 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 July 2, 1981 for a license amendment involving Technical Specification changes which would allow them to repair degraded steam generator tubes by sleeving rather than plugging, which degradation of steam generator tubes had exceeded the plugging limit of 40% nominal wall thickness. In support of this requested change, the licensee has filed with the NRC staff for its review a Westinghouse Steam Generator Report containing technical information regarding tube sleeving of the Point Beach Unit 1 and 2 steam generators. WE modified its application of July 2, 1981 by letter dated October 12, 1981 to request interim operation of Unit 1 with 12 sleeved tubes (no more than six of which have indications of degradation beyond the plugging limit) as a demonstration program until final review of their overall tube sleeving program has been completed.

3.0 SCOPE OF WORK TO BE PERFORMED IN THE DEMONSTRATION PROGRAM

WE has described the scope of the steam generator tube sleeving-demonstration program to be conducted at Point Beach Nuclear Plant, Unit 1 to include the following major steps:

(1) Demonstration of the capability to insert sleeves of two different designs in steam generator tubes with indications of tube degradation. Up to six of these tubes would have degradation in excess of the plugging limit and would include tubes which are presently plugged. The sleeve designs to be used are described in Section 3.2 of Westinghouse Report WCAP-9660 (Proprietary) dated September 28, 1981, and entitled, "Point Beach Steam Generator Sleeving Report for Wisconsin Electric Power Company" (Sleeving Report).

(2) Demonstration and evaluation of the feasibility of explosive and mechanical tube plug removal using plug removal equipment described in Section 4.1 of the Sleeving Report.

(3) Demonstration and evaluation of the tube preparation and sleeving processes and parameters described in Section 4 of the Sleeving Report.

(4) Demonstration and evaluation of the tooling designs required for field installation of sleeves as described in Section 4 of the Sleeving Report.

(5) Demonstration and evaluation of steam generator channel head decontamination equipment described in Section 8 of the Sleeving Report.

(6) Demonstration and evaluation of non-destructive examination techniques described in Section 7 of the Sleeving Report.

4.0 Environmental Impacts Of The Demonstration Program

The Staff has reviewed the radiological and nonradiological environmental impacts of the Demonstration Program. The Staff has identified the radiological environmental impacts of occupational exposure and public radiation exposure as the only measurable environmental impacts of the demonstration program. These impacts are discussed in the following sections.

4.1 Radiological Assessment

4.1.1 Occupational Exposure

We have reviewed the work procedures and practices that Wisconsin Electric Power Company (WE) will use during the steam generator tube sleeving-demonstration project. Based on this review, and through telephone conversations with the licensee, we feel that WE has taken adequate steps to assure that the occupational radiation exposures associated with the tube sleeving-demonstration project will be maintained as low as is reasonably achievable (ALARA) and to assure that the individual doses will be maintained within the requirements of 10 CFR Part 20, "Standards for Radiation Protection".

Wisconsin Electric Power Company (WE) has estimated that the steam generator tube sleeving-demonstration project for the Point Beach Nuclear Plant, Unit 1, will require the expenditure of between approximately 48 and 60 person-rems. The methods used by WE to develop these collective occupational radiation exposure estimates for the steam generator sleeving-demonstration project are based on actual experience and testing. WE 1) determined the maintenance activities that will be involved in the sleeving program; 2) estimated the person-hours of work necessary to perform those activities; 3) determined the areas maintenance personnel must occupy to perform those activities and estimated the radiation dose rates in those areas; 4) multiplied the man-hours by the dose rate for each activity; and 5) summed the doses for all the activities. After reviewing the licensee's methods used to develop those dose estimates, we concluded that these estimates are reasonable. Prior to initiating the steam generator sleeving work, WE will use decontamination techniques in the steam generator channel head area to reduce dose rates. These techniques are expected to reduce the dose rates in the hot leg channel heads of the steam generators by a factor of approximately 2.5[1]. Other ALARA measures implemented by WE during the steam generator sleeving-demonstration project include full size mockups for training workers, use of remote and semi-remote tooling whenever practicable, and routine air sampling, and contamination and radiation surveys. Measures such as these are recommended 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", in order to minimize individual occupational radiation exposures and maintain the overall collective occupational radiation exposure as low as is reasonably achievable (ALARA). No individual will be allowed to exceed the dose limits imposed for workers by 10 CFR Part 20, which are established as dose limits appropriate to the health and safety of individuals.

To determine the relative environmental significance of the estimated maximum occupational dose of 60 person-rems, comparisons were made with 1) the doses expected from normal operation of nuclear plants, and 2) other non-nuclear risks.

Table 4.1 shows the occupational dose history for Point Beach Units 1 and 2[2,3]. When there is more than one reactor unit at a plant site (as at Point Beach) the combined occupational dose for all reactor units (for example, Point Beach Units 1 and 2) can be reported[2,3] instead of the doses for each separate unit. With the addition of 60 person-rems for the sleeving-demonstration project, the average annual dose for the 10 years of dose history at Units 1 and 2 (1970 through 1980) will be approximately 470 person-rems or an average of 235 person-rems per reactor unit. Occupational exposure estimates were not specifically considered in the Point Beach Units 1 & 2 FES[4]. However, in recent environmental statements for new pressurized water reactors (e.g., Summer FES), we have provided an estimate of 410 person-rems per reactor unit as the average annual occupational dose.[5] This estimate is based on reported data from power reactors that are operating with radiation protection programs in accordance with NRC guidance and regulations. A summary of these data is provided in Table 4.2.[2] These data show that 410 person-rems per reactor unit per year is roughly the average of the wide range of doses incurred at all pressurized water reactor units over the last several years. The amount of dose incurred at any single reactor unit in a year is highly dependent on the amount of major maintenance performed that year. Operating data from U.S pressurized water reactors indicates that units requiring high levels of special maintenance work can average as much as 1300 person-rems per year over the life of the unit.[6] Although the doses for these particular plants far exceed the average of 410 person-rems for PWR's, these doses are included in the average and are considered normal deviations from the average, particularly since such maintenance contributes to effective and safe plant operation and since it is carried out with procedures that maintain exposures ALARA. As Table 4.2 shows, the 60 person-rems estimate for the sleeving-demonstration project is within the low end to the historical range of doses for a single unit in a year.

We calculate that 60 person-rems, the occupational dose estimate for the sleeving-demonstration project, corresponds to a risk of very much less than one premature fatal cancer in the exposed work force population. We also calculate that 60 person-rems corresponds to a risk of less than 0.02 genetic effect to the ensuing five generations. These risks are based on risk estimators derived in the BEIR Report[7] and WASH-1400[8] from data for the population as a whole. New information in the BEIR III Report[9] would lead to an even lower estimated risk for premature fatal cancers. These risks are incremental risks (risks in addition to the normal risks of fatal cancer and genetic effects we all face continuously). For a population of 1000, these normal risks that are unrelated to Point Beach Nuclear Station would be expected to result in about 190 cancer deaths and about 60 genetic effects in the existing population (genetic effects are genetic diseases or malformations),[7, 10] plus about 300 more genetic effects among their descendants.

To make the health risk associated with radiation dose more understandable, risk comparisons can be made with non-nuclear activities commonly participated in by many individuals. One rem of radiation is numerically comparable to a lifetime mortality risk of about 10^{-4}.[7] Table 4.3 presents the equivalent risk of 10^{-4} for several common activities - risks which many people take routinely and consider to be insignificant.[11] The average dose to a worker for the sleeving-demonstration project will be roughly 0.6 rems. As Table 4.3 shows, the lifetime risk from radiation dose for the average sleeving-demonstration project worker is smaller than the lifetime risk associated with many common activities.

Another perspective of an occupational risk comes from comparison of occupational mortality risks in the U.S. One such comparison is shown in Table 4.4. It indicates that radiation exposure in the work place, as experienced at an average radiation worker exposure rate, results in a relatively low occupational risk.

Some have criticized occupationally related cancer estimates as being overly conservative.[12] However, most experts feel the risk estimates in Table 4.4 relating to occupational exposure to low-LET radiation are also over-estimates. In our opinion, the comparisons just presented are reasonable ones. The risks of occupational exposures in the range of 0.5 rem per year to 5 rem per year do not significantly affect a typical worker's total risk of mortality.

In summary, the staff has drawn the following conclusions regarding occupational radiation dose. WE's estimate of 60 person-rem for the sleeving-demonstration project at Point Beach 1 is reasonable. This dose is at the low end of the normal range of annual occupational doses which have been observed in recent years at operating reactors. Although the doses resulting from the steam generator tube sleeving-demonstration project will increase the annual collective occupational dose average of Point Beach Units 1 and 2 combined to approximately 470 person-rems, this is still well below the 1300 person-rems per year annual average referenced in current Final Environmental Statements as being an upper bound dose average of PWR's experiencing high levels of special maintenance work. WE has taken appropriate steps to ensure that occupational doses will be maintained within the limits of 10 CFR Part 20 and ALARA. The additional health risks due to these doses over normal risks are quite small, very much less than one percent of normal risk to the project work force as a whole. The risk to an average individual in the work force will be lower than risk incurred from participation in many commonplace activities. The individual risks associated with exposures involved in the sleeving-demonstration program will be controlled and limited so as not to exceed the limits set forth in 10 CFR Part 20 for occupational exposure. For the foregoing reasons, the Staff concludes that the environmental impact due to occupational exposure will not significantly affect the quality of the human environment.

4.1.2 Public Radiation Exposure

NRC Staff has estimated the amount of radioactivity which will be released in liquid and gaseous effluents as a result of the sleeving-demonstration project.[1] Those estimates are presented in Table 4.5. The estimates are based on information supplied by WE[1] to the NRC Staff concerning the method of decontamination and subsequent treatment of the decontamination solutions. Table 4.5 also presents effluent releases for 1979[13] and 1980[14] from Point Beach 1 and the FES[4] annual average effluent release estimates.

WE will take several steps to minimize releases.[1] To minimize airborne releases the channel head decontamination process and the surface preparation process will be wet processes, entraining removed material in water. The air from the channel head where the work is being performed will be exhausted through the opposite manway using a high efficiency particulate filter to control airborne concentrations during channel head work. The water from the decontamination process and the surface preparation process will be treated by filters, an evaporator and a demineralizer to minimize liquid releases.

As Table 4.5 shows, the expected releases from the sleeving-demonstration project are small compared to both the FES estimates and Point Beach's actual annual releases. Therefore, on the basis of this comparison above, we conclude that the offsite environmental impact that may occur during the period of this procedure will be smaller than that which occurs during normal operation.

We have estimated the doses to individual members of the public as well as the population as a whole in the area surrounding Point

Beach Unit 1 based on the radioactive effluents which we estimated for the sleeving-demonstration project (summarized in Table 4.5) and on the calculational methods presented in Regulatory Guides 1.109,[15] and 1.113.[16] Using a liquid release source term of 1.44×10^{-4} Ci consisting primarily of Co-60 (Table 4.5) we calculated the maximum individual total body dose for an adult to be less than .01 mrem for the operations. This is equivalent to a dose of less than a small fraction of 1 percent of the limits of 40 CFR Part 190. The annual limits of 40 CFR Part 190 are 25 millirems to the total body or any organ except the thyroid and 75 millirems to the thyroid. The dose to the population of 819,000[4] within 50 miles was estimated to be less than 6.2×10^{-3} person-rems to the total body from liquid effluents. The offsite population dose was calculated by multiplying the (offsite) maximum individual total body dose of 7.5×10^{-6} mrem (estimated for the liquid release of Co-60) with the projected population of 819,000[4] for the year 1985 within 50 miles of Point Beach 1. We feel that this is a conservative estimate as the maximum individual dose estimate is overly conservative and it is very unlikely that an average individual offsite will receive such a dose. Every year the same population of about 819,000 will receive a cumulative total body dose* of more than 81,900 person-rems from the natural background radiation (about 0.1 rem per year) in the vicinity of Point Beach 1.[11] Thus, the population total body dose from the sleeving-demonstration project is less than 7.6×10^{-6} percent of the annual dose due to natural background. On these bases, we conclude that the doses to individuals in unrestricted areas and to the population within 50 miles due to gaseous and liquid effluents from the sleeving-demonstration project will not be environmentally significant. Since we expect no larger radioactive effluents from Point Beach 1 after the sleeving-demonstration (over presleeving operation), we conclude that the impact on biota other than man will also be no larger than the demonstration project.

In summary, the radioactive releases resulting from the sleeving-demonstration project will be less than those due to normal plant operation. These releases are also much less than the estimates presented in the FES. The doses due to these releases are small

* Our calculations (using the LADTAP Computer Program)[17] for the maximum individual total body dose for an adult considered the following pathway consumption (1) of fish (21 kilogram per year) caught in the discharge area and (2) drinking water (730 liter per year) from the discharge area. A conservative dilution factor of w or no dilution was assumed for each of the above two pathways in our evaluation of radiological exposure due to the release of Co-60 from Point Beach 1 via liquid effluents which are expected to result from the sleeving-demonstration project. The LADTAP II program implements the radiological exposure models described in U.S. NRC Regulatory Guide 1.109, Rev.1 (Appendix a)[15] for radioactivity releases in liquid effluent.

compared to the limits of 40 CFR Part 190 and to the annual dose from natural background radiation. Therefore, the radiological impact of the sleeving-demonstration project will not significantly affect the quality of the human environment.

4.1.3 RADIOLOGICAL ASSESSMENT CONCLUSIONS

Based on our review of the proposed steam generator sleeving-demonstration project, we have reached the following conclusions which are discussed in greater detail above.

(1) The estimated range of 48 to 60 person-rems for the sleeving-demonstration project is on the low side of the expected range of doses incurred at light water power reactors in a year.

(2) The risks to the workers involved in the sleeving-demonstration project from radiation exposure are no larger than the risks incurred by:

  (a) workers in other industrial businesses, and

  (b) most people, working or not, from commonplace activities such as driving a car.

(3) WE has taken appropriate steps to ensure that occupational dose will be maintained as low as it reasonably achievable and within the limits of 10 CFR Part 20.

(4) Offsite doses resulting from the sleeving-demonstration project will be,

  (a) smaller than those incurred during normal operation of Point Beach 1, and

  (b) negligible in comparison to the dose members of the public in the vicinity of Point Beach 1 receive from natural background radiation.

On the basis of the foregoing statements, the staff concludes that the proposed sleeving-demonstration project at the Point Beach

Nuclear Plant, Unit No. 1 will not significantly affect the quality of the human environment.

4.2 Nonradiological Assessment

We have reviewed the documents submitted by WE in support of its request to conduct the steam generator tube sleeving-demonstration program. We find that the proposed activities will occur within the plant on areas previously disturbed during site preparation and construction. These activities will not have appreciable offsite environmental effects. The licensee has not proposed any changes in effluents from the demineralizer waste systems or other waste streams as part of the demonstration program. We conclude that the activities as proposed will not result in any significant environmental impact.

5.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

The NRC has reviewed the Demonstration Program relative to the requirements set forth in 10 CFR Part 51 of the Commission's regulations. The NRC has determined, based on this assessment, that this action will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an Environmental Impact Statement need not be prepared, and that, pursuant to 10 CFR 51.5(c)(1), the issuance of a negative declaration to this effect is appropriate.

TABLE 4.1

ANNUAL COLLECTIVE[2, 3]
OCCUPATIONAL DOSE AT POINT BEACH UNITS* 1, 2

Year	Collective Occupational Dose (person-rems)
1971	164
1972	580
1973	588
1974	295
1975	459
1976	370
1977	429
1978	320
1979	644
1980	791[3]

* First commercial operation 12/70 (Unit 1), 10/72 (Unit 2)

TABLE 4.2

OCCUPATIONAL DOSE AT U.S. LIGHT WATER REACTORS[2]

(person-rems per reactor unit)

Year	PWR Average	BWR Average	Low	High
1969	165	195	42	298
1970	684	127	44	1639
1971	307	255	50	768
1972	464	286	61	1032
1973	783	380	85	5262
1974	331	507	71	1430
1975	318	701	21	2022
1976	460	549	58	2648
1977	396	828	87	3142
1978	429	604	48	1621
1979	510	733	30	2140

TABLE 4.3

LIFETIME MORTALITY RISKS NUMERICALLY EQUIVALENT TO ONE REM[18]

Type of Activity	Equivalent Risk to One Rem
Smoking cigarettes	1 carton
Drinking wine	66 bottles
Automobile driving	6,600 miles
Commercial flying	33,000 miles
Canoeing	1.6 days*
Being a man aged 60	1.8 days

* Eight hours per day

TABLE 4.4

OCCUPATIONAL RISKS

Events per year per 100,000 workers)

	Mining & Quarrying	All U.S. Industries	Trade	Radiation Exposure
Fatal Accidents(1)	63	14	6	1
Delayed Effects				
Actual Observable	readily Observable	Occasionally Observable	not Observable	not Observable
Estimated	?	Includes 115-219 lethal cancers(2)	?	4-6 lethal cancers(3)

---

(1) 1976 data, from "Accident Facts, 1977 Edition," National Safety Council.

(2) Estimates from "Toxic Chemicals and Public Protection, A Report to the President by the Toxic Substançes Strategy Committee," Council on Environmental Quality, Government Printing Office, May 1980. Assumes 20-38% of all cancers are associated with occupation.

(3) Estimates from BEIR-II, 1980, assuming an average radiation worker exposure rate of 0.5 rem/hr; exposure at the limit, 5 rems/yr, would yield an estimate of from 37 to 63 lethal cancers per year per 100,000 workers.

TABLE 4.5

RADIOACTIVE EFFLUENTS FROM POINT BEACH 1

Type of Radioactive Effluent	WE Estimates for Releases During Sleeving Demonstration (Ci)	Point Beach 1 1979 Releases (Ci)	Point Beach 1 1980 Releases (Ci)	FES[1] Estimates of Annual Average Releases (Ci/yr.)
Gaseous				
Noble Gases	Negligible[b]	4.8(+2)[c]	3.2(+2)	5.0(+3)
Iodine + Particulates[a]	Negligible[b]	1.4(-2)	2.7(-3)	1.0(-1)
Tritium	Negligible[b]	4.0(+2)	3.3(+2)	___[d]
Liquid				
Mixed fission and activation products	1.44×10^{-4}	0.38	0.63	1.0(+1)
Tritium	Negligible[b]	4.5(+2)	3.8(+2)	1.0(+3)

[a]Radioactive half lives 8 days or more.

[b]Below lower limits of detectability for plant instrumentation.

[c]4.8(+2) means $4.8 \times 10^{+2}$.

[d]No estimate was given in FES, but FES stated that there would be low concentrations of tritium to the gaseous releases.

References

1. Point Beach Steam Generator Sleeving Report for Wisconsin Electric Power Company prepared by the Westinghouse Electric Corporation, September 28, 1981.

2. NUREG-0713, Vol. 1, Occupational Radiation Exposure at Commercial Nuclear Power Reactors, 1979, U.S.N.R.C., March 1981.

3. NRC Memorandum dated June 19, 1981, from W. E. Kreger to H. R. Denton entitled "Unusually High Occupational Doses Reported For Power Reactors Operating in 1980."

4. Final Environmental Statement related to operation of Point Beach Nuclear Plant, Units 1 and 2, United States Atomic Energy Commission, May 1972.

5. NUREG-0719, Final Environmental Statement Related to the Operation of Summer Pressurized Water Reactor, 1981.

6. NUREG-0692, Final Environmental Statement Related to Steam Generator Repair at Surry Power Station, Unit 1, July 1980.

7. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, "BEIR Report," report of the Advisory Committee on the Biological Effects of Ionizing Radiations, National Academy of Sciences - National Research Council, November 1972.

8. WASH-1400, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S.N.R.C., October 1975.

9. The Effects on Population of Exposures to Low Levels of Ionizing Radiation "BEIR III Report", report of the Committee on the Biological Effects of Ionizing Radiation's National Academy of Sciences - National Research Council, 1980.

10. 1979 Cancer Facts and Figures, American Cancer Society.

11. NCRP No. 45, "Natural Background Radiation in the United States," National Council on Radiation Protection and Measurements, 1975.

12. R. Peto, "Distorting the Epidemiology of Cancer, the Need for a More Balanced Overview," Nature 284, 297-298 (March 27, 1980).

13. Wisconsin Electric Power Company, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Semiannual Monitoring Reports, January 1, 1979 through June 30, 1979 and July 1, 1979 through December 31, 1979.

14. Wisconsin Electric Power Company, Point Beach Nuclear Plant, Unit Nos. 1 and 2, Semiannual Monitoring Reports, January 1, 1980 through June 30, 1980 and July 1, 1980 through December 31, 1980.

15. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (Revision 1), U.S.N.R.C., October 1977.

16. Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," U.S.N.R.C.

17. User's Manual for LADTAP II - A Computer Program for Calculating Radiation Exposure to Man from Routine Release of Nuclear Reactor Liquid Effluents. NUREG/CR-1276, U.S.N.R.C. (May 1980).

18. E. Pochin, "The Acceptance of Risk," British Medical Bulletin 31(3), 1975.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-266

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 56 to Facility Operating License No. DPR-24 issued to Wisconsin Electric Power Company (the licensee), which revised Technical Specifications for operation of Point Beach Nuclear Plant, Unit No. 1 (the facility) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendment is effective as of the date of issuance.

The amendment authorizes power operation of Unit 1 with up to six tubes in one steam generator having degradation exceeding the plugging limit provided these tubes have been repaired by insertion of sleeves to bridge the degraded or defective portion of the tube. It also establishes an additional plugging limit for the six repaired tubes of 35% degradation of the sleeve nominal wall thickness.

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 Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the Federal Register on August 7, 1981 (46 FR 40359). A Petition to Intervene was filed on July 20, 1981 as amended by letter dated August 31, 1981 by Wisconsin's Environmental

Decade. Hearings were held in Milwaukee, Wisconsin on October 29 and 30, 1981. The Board ruled that the NRC staff was authorized to issue the amendment.

The Commission has prepared an environmental impact appraisal of the action being authorized and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action significantly greater than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility dated May, 1972 and the action will not significantly affect the quality of the human environment. Therefore the Commission has determined that the issuance of a negative declaration to this effect is appropriate.

For further details with respect to this action, see (1) the application for amendment dated July 2, 1981 as modified by letter dated October 12, 1981, (2) Amendment No. 56 to License Nos. DPR-24, (3) the Commission's related Safety Evaluation, (4) the Commission's Environmental Impact Appraisal and (5) the Atomic Safety and Licensing Board's Memorandum and Order dated November 5, 1981. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Joseph Mann Library, 1516 16th Street, Two Rivers, Wisconsin 54241. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 10th day of November, 1981

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing