DCS MS-016

~~Docket File~~ ASLAB
NRC PDR
Local PDR
ORB Rdg #3
DEisenhut
PMKreutzer
OELD
SECY
LJHarmon (2)
TBarnhart (8)
LSchneider (1)
DBrinkman
ACRS (10)
OPA(Clare Miles)
RDiggs
~~RBallard~~
NSIC

APR 1 8 1983

Docket Nos. 50-266
and 50-301

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

Dear Mr. Fay:

Pursuant to the Initial Decision of the Atomic Safety and Licensing Board dated February 4, 1983, on April 4, 1983 the Commission issued Amendment Nos. 71 and 76 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments were effective immediately and consisted of changes to the Technical Specifications to allow repair of degraded steam generator tubes by sleeving which would otherwise be required to be plugged and removed from service; established primary coolant limits for iodine concentration and surveillance frequency, and established a plugging limit for sleeved tubes of 40% nominal sleeve wall thickness.

On April 13, 1983, the Resident Inspector called to inform us that his copy of the amendment did not contain Technical Specification pages. Upon further checking with your offices and the plant, it was determined that, as of April 18, 1983, the amendment had not been received by your office.

In order to avoid further problems and inconveniences, the entire Amendment is hereby reissued. Please accept our apologies for this clerical error.

Sincerely,

Original signed by:

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

Enclosure:
April 4, 1983 letter with
attached Amendments and
safety evaluation.

cc w/enclosures
See next page

8304260097 830418
PDR ADOCK 05000266
P PDR

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL				
SURNAME	PKreutzer	TColburn:dd	RAClark				
DATE	4/18/83	4/18/83	4/18/83				

Wisconsin Electric Power Company

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

USNRC Resident Inspectors Office
6612 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. James J. Zach, 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

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

Regional Administrator
Nuclear Regulatory Commission, Region III
Office of Executive Director for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

Docket Nos. 50-266
and 50-301

APR 4 1983

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

Dear Mr. Fay:

Pursuant to the enclosed Initial Decision issued by the Atomic Safety and Licensing Board dated February 4, 1983 (ASLBP No. 81-464-05 LA) we have issued the enclosed Amendment Nos. 71 and 76 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments are effective immediately and consist of changes to the Technical Specifications to allow repair of degraded steam generator tubes by sleeving which would otherwise be required to be plugged and removed from service; establish primary coolant limits for iodine concentration and surveillance frequency, and establish a plugging limit for sleeved tubes of 40% nominal sleeve wall thickness.

Steam generator tube sleeving was the subject of a hearing presided over by the Atomic Safety and Licensing Board (Board). The Board stipulated certain conditions in its Initial Decision concerning the litagable issue which survived summary disposition (adequacy of eddy current testing). These conditions reflect understandings of the hearing record. These conditions are that:

a. Steam generator tubes that have been previously subject to explosive plugging shall not be sleeved;

b. Brazed joints shall not be employed;

c. Should eddy current testing indicate 40 percent or more degradation from the nominal tube wall thickness of a sleeve, the sleeved steam generator tube shall be plugged; and

d. Leak limits previously imposed on the repaired steam generators shall continue to apply.

Your Technical Specifications as originally proposed July 2, 1981 and amended March 9, 1983 have been modified, as discussed with your staff, to include conditions found in the Board's Initial Decision. Namely, brazed joints shall not be employed, tubes that have been previously subject to explosive plugging shall not be sleeved and the plugging limit for sleeve wall degradation shall be 40% nominal wall thickness.

8304130477

The TS have also been modified to correct minor discrepancies between the proposed TS and the Standard TS for Westinghouse Pressurized Water Reactors.

The evaluation regarding the establishment of primary coolant activity limits for iodine concentration and surveillance frequency consistent with the Standard Technical Specifications for Westinghouse Pressurized Water Reactors is contained in the NRC staff's Safety Evaluation dated July 8, 1982. This document was introduced as a part of the hearing record.

It should be noted that Wisconsin's Environmental Decade, the intervenor in the hearing, filed an appeal to the Board's Initial Decision by letter dated February 14, 1983.

A copy of the related Notice of Issuance which is being filed with the Office of the Federal Register for Publication is also enclosed.

Sincerely,

Robert A. Clark

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

Enclosures:
1. Initial Decision dated
   February 4, 1983
2. Amendment No. 71 to DPR-24
3. Amendment No. 76 to DPR-27
4. Notice of Issuance

cc: w/enclosures
See next page

Enclosure 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
Peter B. Bloch, Chairman
Dr. Jerry R. Kline
Dr. Hugh C. Paxton

In the Matter of	Docket Nos. 50-266-OLA 50-301-OLA
WISCONSIN ELECTRIC POWER COMPANY	ASLBP No. 81-464-05 LA
(Point Beach Nuclear Plant, Units 1 & 2)	February 4, 1983

MEMORANDUM AND ORDER
(Initial Decision)

This decision concerns the adequacy of eddy current testing to detect potentially serious defects in corroded steam generator tubes that have been repaired by the insertion of a liner or "sleeve."[1] The "sleeve" is designed to lend structural strength to the tube by spanning its corroded area.[2]

We have found limits in the capability of the eddy current test to detect flaws in steam generator tubes. However, we have concluded that

---

1 This is the only issue remaining in the proceeding because we granted summary disposition of the rest. LBP-82-88, 15 NRC ____ (October 1, 1982)(Summary Disposition).

2 On July 2, 1981, Wisconsin Electric Power Company (applicant) filed a Technical Specification Change Request, seeking to amend the Point Beach Operating licenses to permit repair of steam generator tubes that have degradation exceeding 40% of the nominal tubewall thickness. The existing plant Technical Specifications require that such tubes be removed from service by "plugging." The proposed Technical Specification change would permit repair of such tubes by "sleeving," leaving the tubes in service.

8302070506

these limits of eddy current testing do not seriously detract from its ability to detect flaws that are likely to rupture, either under normal operating conditions or postulated accident conditions. Furthermore, sleeved tubes appear to be safer than other unsleeved tubes that applicant already is licensed to operate. We also have concluded, based on an analysis of various factors affecting the safety of sleeves, that sleeved tubes are safe, without reference to whether they are safer than unsleeved tubes. Consequently, the license amendment should be granted, without any conditions attached at the direction of the Atomic Safety and Licensing Board.

I. DESCRIPTION OF SLEEVING

In order to understand the nature of the problem that gave rise to the issues in this case it is useful to describe briefly the functions of a steam generator in a nuclear power plant.[3] All pressurized water nuclear power plants, including the Point Beach units, have two systems of piping to effect the transfer of energy from the reactor core to the turbines which produce electricity. The primary system pumps circulate primary coolant water around the hot fuel rods within the reactor core where the nuclear reaction takes place. The super-heated water then passes through large pipes to the steam generators. In each steam generator -- heat exchangers approximately 70 feet high and fourteen

---

[3] The general description of the role of a steam generator is taken from Florida Power & Light Company (Turkey Point Nuclear Generating, Units Nos. 3 and 4), ALAB-660, 14 NRC 987 (1981) at 992.

feet in diameter -- the primary coolant water passes from large pipes into about 3000 smaller tubes which are partially immersed in a separate system of water, the secondary coolant. Heat is transferred through the tube walls from the primary coolant to the secondary coolant, which boils and, in the form of steam, passes through turbines to generate electricity. In order to prevent leaks of primary coolant and radioactivity from the primary system to the secondary coolant, it is necessary to assure the integrity of the entire piping system, including each of the thousands of small tubes inside each steam generator.

At Point Beach, steam generator tubes have experienced substantial thinning and corrosion, caused initially by the use of a phosphate chemistry regime in the secondary side water but continuing to some degree even after the secondary side chemistry was changed to an "all volatile" chemistry regime. As a result, applicant sought to repair these degraded steam generator tubes and, on July 2, 1981, filed a Technical Specification Change Request, seeking to amend the Point Beach operating licenses to permit repair of steam generator tubes that have suffered from corrosion. Without the amendment, applicant would have to remove from service (by plugging both ends of the tube) all tubes that have been degraded by more than 40% of their design (or "nominal") tubewall thickness.

The repair consists of the insertion of a liner or "sleeve" into the degraded tube, spanning the area where the corrosion has occurred. Then the sleeve is joined at its top and bottom to the exterior tube.[4]

There are two steam generators at each of the Point Beach units. Each steam generator contains 3260 inverted, U-shaped vertical tubes. The ends of the tubes pass through and are anchored in the tubesheet. The tubesheet is a large circular steel plate, about 22 inches thick, through which holes are drilled for the tubes. The bottom 2½ to 3 inches of the end of each tube is fastened within the bottom of the tubesheet by "rolling," i.e., the tube is mechanically expanded tightly against the walls of the tubesheet hole. The tubes are also welded at the bottom face of the tubesheet. The tubes are not fastened at the top of the tubesheet.[5]

The sleeving process involves the insertion of a smaller diameter, thermally treated Inconel 600 metal sleeve inside a steam generator tube so that the bottom of the sleeve is flush with the bottom of the tube. The sleeve extends beyond the top of the tubesheet, bridging the degraded portion of the tube. The sleeve is bonded to the tube at the bottom and just below the top of the sleeve.[6]

---

4 See LBP-81-55, 14 NRC 1017 (1981) at 1019.

5 Affidavit of David K. Porter (September 28, 1981) at ¶4 (Attachment 1 to "Licensee's [applicant's] Motion for Authorization for Interim Operation of Unit 1 With Steam Generator Tubes Sleeved Rather Than Plugged," September 28, 1981). (Porter Affidavit.)

6 The sleeve is designed to extend beyond its upper joint so that the additional length of sleeve would prevent a failure of the upper joint

II. COMMENTS ON THE "STATEMENT OF INADEQUATE RECORD"

Wisconsin's Environmental Decade (Decade), the sole intervenor, did not present any witnesses, attempting to rely on cross-examination to establish its case. It also did not file formal findings pursuant to the Board's request.[7] Instead, it filed a five page "Statement of Inadequate Record." That document contains a few relevant and helpful points, but it was a disappointment to the Board because it failed to provide us with any reasoning by which we could dispose of the litigated issue in Decade's favor.[8]

---

from resulting in an unconstrained rupture. Should the joint fail, the sleeve will remain within the tube, restricting the amount of water that can leak through the joint area. Porter Affidavit at ¶5; Applicant Exhibit 1, § 3.2.

[7] Tr. 18767-78.

[8] Decade's Statement of Inadequate Record urges the Board to conduct what is essentially a probabilistic risk analysis for steam generator tube burst. Such an analysis would assess the overall risk to public health and safety by considering both the probability of tube burst and the consequences of that event.

In this proceeding the Board has not undertaken such a quantitative analysis, using fault trees, numerical probabilities of failure of components and numerical estimates of overall risk. The Board nevertheless considered, in its Summary Disposition decision, what its course might be should eddy current testing prove to be inadequate for the detection of flaws in sleeved tubes. It therefore requested the applicant and staff to address contingently the safety implications of sleeving if that finding was made. Both did so. We consider those implications in subsequent sections of this decision even though we could rest our decision solely on the demonstrated adequacy of eddy current testing. The record therefore does reflect thorough consideration of both the likelihood of not finding flaws and the consequences of not finding them. Of course, we do not use the format of probabilistic risk analysis, which is not required by Commission policy or regulations.

Decade attempts to excuse its Statement on the grounds that it was required to work during the Christmas vacation. However, Decade failed to request a time extension, either during the hearing or in its filing. Furthermore, we know that Decade is aware that it can obtain extensions of time limits for good cause, as it was permitted to file its Motion for Litigable Issues after the time originally set.

Although Decade's filing is a disappointment to us, we do not assess any sanctions against it, primarily because we "requested" the filing of findings but never thought it necessary to order that they be filed. The result is that we will do our best to respond to the few arguments Decade has made and to analyze the validity of the case presented to us in the briefs of the other parties. We are pleased with briefs filed by applicant and by the Staff of the Nuclear Regulatory Commission (staff), which respond well to our requests for a reasoned discussion of the entire record.

III. ANALYSIS AND CONCLUSIONS

In this section of our opinion, we discuss the contention that was admitted to the hearing, the applicable regulatory materials, the facts concerning the reliability of eddy current testing, and the redundant protections from steam generator tube failure available at Point Beach.[9] Appendix A lists our previous decisions in this proceeding.

---

[9] To simplify our discussion, we include a list of our previous decisions in Appendix A and a brief statement of the qualifications of each of the witnesses in Appendix B. We consider each of the witnesses

A. The Admitted Contention

This contention, as originally submitted, was quite lengthy and was intertwined with other assertions. The contention was:

> Present inspection methods [understood to be limited to eddy current testing[10]] in unsleeved tubes have been shown to be inadequate to detect defects, and the complicating presence of the sleeve inside the tube will make the detection of degradation, especially at the joints, even more difficult. Over time, the detection capability will continue to degrade. . . . The inability to adequately detect defects that can lead to primary-to-secondary or secondary-to-primary pathways for leakage will exacerbate the problems indicated in [the other subissues in this allegedly litigable issue.][11]

However, our Summary Disposition decision modified this contention by determining that the following genuine issue was admitted to hearing:

> That the license amendment should be denied or conditioned because applicant has not demonstrated that eddy current testing is adequate to detect serious stress corrosion cracking or intergranular attack, in excess of the technical specification prohibiting more than 40 percent degradation of the sleeve wall, in sleeves that would be inserted within steam generator tubes.[12]

---

to be an expert.

10 Tr. 1237-38.

11 See Summary Disposition 15 NRC ____ (October 1, 1982), slip op. at 10.

12 Id. at 1.

This admitted genuine issue was discussed in our Summary Disposition decision in some detail, explaining what issues of fact or opinion the Board considered unresolved.[13]

B. Regulation Involved

The Nuclear Regulatory Commission (NRC) regulation covering the adequacy of eddy current testing relates generally to the design of the reactor coolant pressure boundary. That regulation, General Design Criterion 14, Appendix A, 10 C.F.R. Part 50, requires that:

> The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

In order to comply with this General Design Criterion, applicant's proposed repair proposal adheres to an industry code, the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code (Code)[14].

C. Adequacy of Eddy Current Testing

In this section of our opinion, we will describe eddy current testing (ECT) and then evaluate its reliability for detecting leaks.[15]

---

13 Id. at 2, 10-16.

14 Licensee Exhibit 1, §3.1.

15 We have leaned heavily on applicant's Proposed Initial Decision, 17-20, for this portion of our decision.

1. Description of Eddy Current Testing

For ECT, a probe is inserted into the steam generator tube. Electric current within the coils in the probe produces an electromagnetic field. As the probe is moved within the tube, an electric current is induced in the conductive material of the tube or sleeve. This is the eddy current signal that is recorded and interpreted. Degradation in the wall of the tube or sleeve causes variations in the effective electrical conductivity or magnetic permeability of the wall material. These variations are measured directly by changes in the coil voltage of the eddy current probe.[16]

ECT at Point Beach is performed by Westinghouse Electric Corporation, which subcontracts the reading and interpretation of the eddy current data to Zetec, Inc.[17] Mr. Denton and Mr. McKee, of Zetec, offered testimony in considerable detail about ECT equipment, the physics of the ECT process, the interpretation of eddy current signals, and the capabilities of ECT for detecting, in the field, stress corrosion cracking (SCC) and intergranular attack (IGA) in tubes and sleeves.[18]

---

16 "Licensee's [Applicant's] Testimony of W.D. Fletcher" (Fletcher), ff. Tr. 1422, at 3-4; Tr. 1462-64, testimony of Clyde J. Denton (Denton).

17 Tr. 1460-61 (Denton).

18 Tr. 1462-78 (Denton); Tr. 1608-1723 (Denton, McKee); Applicant Exhibits 2 and 3. IGA is corrosion of the metal grain boundaries of the tube material that does not initially result in separation of the metal grains. SCC entails distinct separation of the metal grains resulting from corrosion. Tr. 1427-31 (Fletcher).

The eddy current signals for each tube that is tested are recorded on a magnetic tape. The tape is used to produce a strip chart which converts the record of electromagnetic signals into a linear graph that roughly resembles the record of an electrocardiograph. This chart indicates the presence or absence of defect signals along the tubewall.

If the strip chart indicates that degradation may be present,[19] the magnetic tape recording of the eddy current signals also is used to generate a picture on an oscilloscope. That moving picture is recorded in a still photograph that enables the operator to examine phase differences between signals coming from the outside and inside tube surfaces. That still photograph is then interpreted to determine the depth of penetration of degradation into the tubewall material.[20]

An eddy current indication of a defect in the tubewall appears as a deviation from a base line drawn along the center of the strip chart. The greater the volume of the defect, the greater the amplitude of the deviation from the base line.[21] Unwanted signals, or "noise," also appear as deviations from the base line on the chart. Noise is caused by such extraneous sources as conductive impurities deposited on the surface of the tube, magnetite in sludge surrounding the tube, or the

---

[19] Tr. 1658-1659.

[20] Tr. 1608-11; 1473 (Denton).

[21] Tr. 1611, 1620 (Denton).

uneven inner surface of a structure surrounding the tube--such as the inner surface of the tubesheet hole.[22]

An important concept used in diagnosing potential defects is the "signal to noise ratio." This is the ratio of the amplitude of the signal generated by a suspected defect to the amplitude of the noise signals found in the same general region of the strip chart. Multifrequency mixing techniques are used to significantly reduce the amplitude of the noise signals.[23]

The amplitude of the eddy current signal is indicative of the volume of the degradation, meaning the amount of separation present in the tubewall; but the amplitude says nothing about the depth of penetration into the tubewall.[24] When the eddy current interpreter sees a signal which might indicate degradation, the signal is examined on the oscilloscope.[25] When signal-to-noise ratios are less than about three-to-one, operators must exercise substantial judgment about whether or not a defect exists and whether the investigation should be pursued further by reading the signal on the oscilloscope.[26]

---

[22] Fletcher, ff. Tr. 1422, at 4.

[23] Fletcher, ff. Tr. 1422, at 4; Murphy, ff. Tr. 1828, at 8; Staff Exhibit 1, at 32.

[24] Tr. 1611 (Denton); Tr. 1495-96 (Fletcher); Tr. 1672 (Denton).

[25] Tr. 1473, 1610 (Denton); Tr. 1631 (McKee). The voltage of the pattern displayed on the screen, or "voltage lissajous," also provides a rough indication of the volume of the defect. Tr. 1657-58 (Denton).

[26] Tr. 1649-50 (Denton).

When a photograph of the oscilloscope picture is made, the duration of the exposure is sufficient to depict the two phases of the oscilloscope pattern that are of concern. A picture of the oscilloscope pattern of a crack in a tubewall would typically appear on the scope in the shape of a flattened figure eight.[27]

The angle between the two significant phases of the oscilloscope picture, as measured with an electronic protractor, indicates the depth of the penetration.[28] For defects of very small volume, the figure on the scope may be small, and the phase angle may be difficult to measure precisely. In such cases, the interpreter is expected to take the most conservative reading of the angle, thus tending to overstate the depth of penetration.[29]

Under Board questioning the staff stated that they would require a tube to be plugged if the indicated depth of penetration exceeded 40% even under circumstances where the degree of penetration was reported conservatively (i. e., the true penetration was likely to be less than 40%).[30]

---

27 Tr. 1471-73, 1618-20 (Denton); Applicant Exhibit 2, at 1; Applicant Exhibit 3.

28 Tr. 1611-12, 1677 (Denton).

29 Tr. 1622 (Denton).

30 Tr. 1855-56 (Murphy).

2. Reliability of Eddy Current Testing

The reliability with which eddy current testing detects corrosion flaws depends on the volume of the flaw[31] in the steam generator tubewall and not on the depth of penetration of the flaw into the tube. This detracts somewhat from the utility of the test since it is the depth of penetration which is the principal variable of interest for licensing; NRC technical specifications require that a tube be plugged when a flaw penetrates the tubewall by 40 percent or more of the wall thickness.

The volume of the flaw is, however, related indirectly to the depth of penetration. Experience indicates that cracks propagate through the tubewall with aspect ratios having a value of about two to five. (The aspect ratio is the ratio of the length of a crack on the outside surface to the depth of penetration.) Thus, field experience shows that cracks in tubes which could be of significance to NRC enforcement of its plugging limits have in most (but not all) instances adequate volume to be detected by eddy current testing.[32]

One expert testified that for a flaw with sufficient volume to be detected (i. e., the signal to noise ratio is greater than about 3) a

---

[31] The volume of the flaw is the volume separation in the tubewall or the amount of material that could hypothetically be inserted into the flaw See Tr. 1695-96.

[32] Fletcher, ff. Tr. 1422, p. 3, 7-8; Murphy, pp. 8, 9.

A penetration of the wall might not be detected, for example, if it has a shape analogous to a small diameter drill hole of small volume. Tr. 1691 (Denton).

50 percent wall penetration can be measured with precision (test-retest reliability) of about ± 7 percent. The precision diminishes as the crack size diminishes (*i. e.*, the error increases) so that a 30 percent through-wall crack could be measured with a precision of about ± 13 percent.[33]

The likelihood of detection of a crack (as opposed to the precision with which it can be measured) is about 95 percent certainty for a 40 percent penetration having a 150 mil axial surface crack length. A similar crack having only 20 percent penetration might not be detected at all.[34]

The limits of usefulness of eddy current testing are known. Eddy current testing using bobbin type coils cannot be used to detect circumferential cracks in tubes since the lines of current flow are parallel to such a crack and are therefore not interrupted as they are by axial cracks which are oriented normal to the electric field.[35] However, the mode of cracking generally found is axial because of hoop stresses in the tube. In fact, circumferential cracks have not been found at Point Beach.[36]

The technique also cannot be relied upon at present to detect intergranular attack (IGA) which is unaccompanied by cracking. This is

---

[33] Tr. 1690-92 (Denton).

[34] Tr. 1695 (McKee).

[35] Murphy, 8, 9.

[36] Fletcher, ff. Tr., p. 1740.

because the current flow from the probe is not interrupted by IGA alone; the uncracked tube material continues to act as an electrical conductor even though it is corroded. Separation of grain boundaries through cracking is needed for detectability. This has proven to be of significance for locations within the tubesheet where enough sludge has accumulated in the crevice between the tubes and tubesheet wall to prevent separation of grain boundaries in corroded tubes. Tubes leaking within the tubesheet have occasionally not been found by eddy current testing because of this phenomenon.[37]

Eddy current testing alone cannot be relied upon for diagnosis or detection of corrosion over its full range of possible occurrence. Physical parameters such as interference (from magnetite or copper in sludge), variations in the tube diameter, machine marks, denting in tubes, and small flaw volumes impose limits on detectability.[38] As a practical matter this suggests that leaking tubes occasionally will not be detected by eddy current testing.[39]

The instances where eddy current testing failed to detect either penetrations exceeding the plugging limit or actual leaking tubes are attributable to the flaws being at or below the physical limits of detection. This may occur because of interference of the signal, the

---

37 Murphy, ff. Tr. 1828, pp. 5,6.

38 Fletcher, p. 4.

39 Fletcher, p. 6.

small volume of the defect or the constraining effect of sludge within the tubesheet.

The board concludes, however, that the applicant, its consultants and the NRC staff are familiar in detail with the inherent physical limitations of the eddy current technique for detecting stress corrosion cracking. Applicant does not rely, for safety, on eddy current measurements that are outside of the inherent bounds of reliability of the instrument.

The principal safety related use for eddy current testing in steam generators is for enforcement of NRC's 40 percent plugging limit, which is conservative because it takes into account uncertainties of measurement. Analyses show that uniform thinning completely around the circumference of the tube to 62 percent degradation would not result in tube rupture following a main steam line break. Burst tests on tubes having 40 to 60 percent through wall-penetrations confirm that burst would not occur even at pressures anticipated in a main stream line break.[40]

The purpose for setting plugging limits and for inspection of tubes is to prevent corrosion of tubes to progress undetected to the point where rupture is likely under either accident conditions or normal operation.[41] It is particularly important to safety to have the capability for detecting relatively large volume defects (those above

[40] Fletcher, ff. Tr. 1422, p. 9; Murphy, pp. 3-4.

[41] Fletcher, p. 10; Murphy, p. 3.

the plugging limit) so that tubes can be plugged before a hazardous condition arises.

Much was made at hearing about the uncertainties attendant to the lower limits of detection for eddy current testing, where it is beyond question that the technique does not detect every small flaw.[42] While it was necessary to probe those limits, we now conclude that the limits of detection inherent to eddy current testing do not cause a concern that stress corrosion cracking could progress undetected to the point that large tube rupture from that mechanism is at all likely.[43]

---

[42] Eddy current testing failed to detect the source of a known leak in one steam generator tube, and it is not unusual for a through-wall defect to appear on an eddy current test to be an 80 percent defect Tr. 1661-64 (Denton). Additionally, an eddy current test sometimes has shown a defect as great as 90 percent that was not detected at all in testing conducted just six months before. Tr. 1643-47 (Denton). This indicates a high degree of uncertainty in these particular readings because reliable laboratory tests conducted on samples of mill annealed Inconel 600 indicate that the maximum rate of deterioration in a highly caustic environment during a six month period was no more than 7.5% Fletcher, ff. Tr. 1422 at 6.

These limitations in eddy current testing are known. Since 1979, Westinghouse has conducted research to improve the early detection of IGA. Recently, Westinghouse has developed a process for exposing tubing to an acid condition to produce laboratory samples with IGA of various depths of penetration, unaccompanied by cracking. Westinghouse is testing the eddy current response to the IGA which, rather than the relatively sharp deviation caused by an SCC signal, is a "drift" from the base line on the strip chart. On an experimental basis, it now seems possible to detect 20% wall penetration by IGA in the laboratory; and work is continuing to develop a standard that will enable the interpreter to recognize IGA in the field. Tr. 1437-47 (Fletcher).

[43] Murphy, pp. 7-8.

3. Detecting Flaws in Sleeves

To this point, we have discussed difficulties in using eddy current testing in any tube in a steam generator. However, a narrower question rests before us. Applicant is licensed to operate its plant according to its existing technical specifications. It may operate any tube in its steam generator until eddy current tests show 40% or more degradation of the nominal tubewall thickness. At that point, the technical specifications require the tubes to be plugged. Our jurisdiction is to decide whether it is safe to operate those degraded tubes with sleeves rather than plugs. We have no jurisdiction over the safety of the remainder of the steam generator, which applicant already is licensed to operate.[44]

We conclude that the sleeving process reinforces and strengthens existing steam generator tubes. No serious question has been raised about the integrity of the joints by which the sleeves are bound to the existing tubes. The result is that, at the time the sleeves are inserted, the new and undegraded sleeve replaces the degraded tube as a portion of the primary pressure boundary of the reactor. At that time, the sleeve enjoys greater integrity than many of the degraded tubes that applicant already is permitted to utilize in its steam generator.

---

[44] See Northern Indiana Public Service Company (Bailly Generating Station Nuclear 1), ALAB-619, 12 NRC 558, 565 (1980); Public Service Company of Indiana, Inc. (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-316, 3 NRC 167, 171 (1976).

Furthermore, this new primary pressure boundary is made of a corrosion resistant material, thermally treated Inconel 600, which is two to three times more resistant to corrosion than the initial steam generator tubes,[45] which were not thermally treated to enhance their corrosion resistance.

The safety of the newly installed sleeves may be further enhanced if ongoing research succeeds in improving the ability to detect corrosion using eddy current testing.[46] This would permit corrective action.

Even if ongoing research does not succeed, sleeved tubes will be safer than unsleeved tubes. To the extent that there may be imprecision in the tests currently in use in steam generator tubes, including eddy current testing and hydrostatic testing[47], the insertion of new sleeves provides a margin of comfort not found in other tubes. The other tubes, which have been used for many years, are subject to undetected corrosion; the new sleeves will take many years before their exposure to the steam-generator environment might cause an analogous risk in them.

---

[45] Corrosion resistance of thermally treated Inconel 600 has been tested in the laboratory. IGA was shown to have been reduced by two to three times and stress corrosion cracking by about ten times. Fletcher, ff. Tr. 1422, at 6-7; Murphy, ff. Tr. 1828, at 2; Tr. 1483-88 (Fletcher).

[46] Tr. 1437-47 (Fletcher).

[47] Discussed below.

Sleeves also will initially confront a less hostile environment than will existing tubes. Most sleeves will be protected from the secondary-side environment by the tubes into which they are inserted. They will be exposed to the secondary side only if the repaired tube develops a substantial leak, thus permitting the potentially corrosive materials in the secondary side to touch the sleeve.[48]

Although neither applicant nor staff depends on the presence of the tube around the sleeve to support its belief that the sleeved tubes have an adequate safety margin, it is obvious that the presence of the tube enhances the safety of the sleeve. If the sleeve were to rupture, it is possible that the surrounding tube would be so degraded that it would in no way constrain the resulting leak. However, it is likely that the degradation of the tube would be in a different region than the rupture in the sleeve. In that case, the intact tube may constrain both the rupture and the leak from the sleeve. While there is no assurance that this constraint would occur, this possibility weighs on the side of greater safety for a sleeved than for an unsleeved tube.[49]

An interesting beneficial side-effect of sleeving is that it will retard the process of corrosion of the surrounding tube. This will occur because the sleeve will somewhat insulate the tube from the heat

---

[48] Fletcher, ff. Tr. 1422, at 6.

[49] See Marsh, ff. Tr. 1822, at 3-4; Murphy, ff. Tr. 1828, at 4.

of the primary system. This reduction in temperature should be accompanied by a reduced rate of corrosion, which is facilitated by heat.[50]

It is also likely that the thermal-hydraulic properties of the tube-sleeve annulus[51] will retard the accumulation of corrosive materials. The most likely pathway for leakage into the annulus would be through the tubewall near the top of the tubesheet; this is the area of the steam generator where the greatest corrosion has occurred.[52] The sleeve, in direct contact with the heated and pressurized primary coolant, will turn the water in the annulus to steam, which will escape through the leakage pathway from which it entered.[53] Consequently, the turnover of water and the deposition of sediment in the annulus would be severely limited,[54] retarding the rate of accumulation of corrosive materials in the annulus, as compared to the accumulation at the top of the tubesheet. The result is that there would be less

---

[50] Murphy, ff. Tr. 1828, at 2; Tr. 1769-70 (Fletcher); Tr. 1851, 1859-60 (McCracken).

[51] The space between the tube and sleeve is known as the "annulus."

[52] Fletcher, ff. Tr. 1422, at 10; Tr. 1767-69 (Fletcher); Tr. 1851 (McCracken).

[53] Mr. Fletcher anticipated that only a small amount of water would enter the annulus before flashing to steam. Ordinarily, this would be the case. However, as corrosion progresses a substantial amount of water could leak into the annulus during a period of cold shutdown. It is our conclusion that steam still would form when the generator was returned to service following such a period, so we accept the implications of Mr. Fletcher's analysis for the slightly different hypothetical situation we have envisioned. Tr. 1766-73 (Fletcher); Tr. 1851-52 (McCracken); Tr. 1853 (Murphy).

[54] _See_ Tr. 1769-71.

sediment to facilitate corrosion of the sleeve, as compared to the amount of sediment facilitating corrosion of an unsleeved tube. Hence, the sleeved tube should be subject to a slower rate of corrosion.

Finally, we conclude that whatever the difficulties of eddy current testing, it is a more accurate instrument for testing the sleeve (below the upper joint) than for testing unsleeved tubes. (We do not examine questions concerning the upper joint because we previously found there was no genuine issue of fact concerning the testing of the upper joint.[55]) The principal reason for increased inspectability is that noise from the tubesheet crevice will be reduced because the sleeve is separated from the crevice by the thickness of the surrounding tube plus the width of the annulus between the tube and sleeve.[56] The outer surface of the sleeve is 75 mils away from the surface of the tubesheet hole. This significantly reduces the noise level.[57]

In summary, we find that sleeved tubes are safer than unsleeved tubes already present in the Point Beach steam generator. In addition, these tubes are easier to inspect for degradation that may occur. Hence, we conclude that the sleeved tubes will be subject to an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture[58] and that we should approve the request to amend

---

55 Summary Disposition, slip op. at 15.

56 Fletcher, ff. Tr. 1422, at 3-5.

57 Id. at 4.

58 General Design Criterion 14, Appendix A, 10 CFR Part 50.

applicant's operating license to permit the sleeving of tubes that otherwise would be required to be plugged.

D. Safety Factors in Sleeved Tubes

The safety of sleeved tubes does not depend on eddy current testing alone. Consequently, although the admitted contention deals with eddy current testing, our Summary Disposition decision invited evidence concerning the relationship between the testing program and the safety of the reactor.[59] In response, evidence was submitted that persuades us that protection from steam generator tube failures depends on a series of safety factors, including:

1. Design, fabrication and testing in compliance with the ASME Boiler and Pressure Vessel Code.
2. Hydrostatic testing
3. Continuous leak monitoring
4. Leak-before-break characteristics of tubing material
5. Conservative criteria for utilizing eddy current test results
6. Possible leak constraint from the presence of the tube around the sleeve or from the tubesheet, and
7. Likelihood of a less corrosive environment within the sleeve-tube annulus.

In this section of our opinion, we shall discuss each of these safety factors. Although we could rest our opinion solely on the conclusions we reached above concerning the increased safety of sleeved tubes,

---

[59] _See_ Summary Disposition, slip op. at 14.

compared to unsleeved tubes, we also conclude that the combined effect of these seven factors contributes to safety, thereby complying with General Design Criterion 14. Our review of these safety factors also persuades us that it would not be appropriate for us to initiate an inquiry of our own into possible safety or environmental problems with the sleeving project.[60]

1. Compliance With ASME Code and Additional Testing

Steam generators, including the tubes and sleeves, are designed, fabricated and tested in accordance with design criteria which include compliance with the ASME Boiler and Pressure Vessel Code.[61] To further assure itself of the safety of the proposed sleeving repair process, applicant had Westinghouse Electric Corporation conduct extensive analyses and laboratory tests.[62] The ensuing "Sleeving Report" contains results of a design verification test program whose objective was to assess the structural integrity and corrosion resistance of

---

60 Atomic Safety and Licensing Boards have the authority to pursue relevant safety and environmental issues that arise in the course of a proceeding. 10 CFR §2.760a. Although the use of this "sua sponte" authority has been made dependent on Boards first notifying the Commission of their action in declaring a sua sponte issue, the continued existence of the authority to declare such issues imposes on Board the responsibility of considering whether or not to declare such issues. Although it may not be strictly necessary to explain why that authority has not been exercised, this Board believes it preferable to expose its decisional process to public scrutiny.

61 Applicant Exhibit 1, §3.1.

62 Westinghouse Electric Corporation, Point Beach Steam Generator Report, September 1981 (Revised February 1982)(Sleeving Report).

sleeved tubes.[63] The laboratory tests that were performed included a variety of corrosion and structural tests on tube materials and on sample tubes.

At an earlier stage of this proceeding, we addressed a limited number of questions to the applicant concerning possible problems in the Sleeving Report. As a result, we satisfied ourselves that the Sleeving Report was prepared with reasonable care and we were unable to identify any serious deficiencies for us to pursue. At this stage of the proceeding, the Sleeving Report also provides us with assurance that the sleeving project was carefully designed and tested and that there are no important safety or environmental issues for us to pursue.

Sleeved tubes will have greater integrity than unsleeved tubes. The sleeves are made of thermally treated Inconel 600, which has greater resistance to corrosion than the mill annealed Inconel 600 used in the original tubes. Laboratory tests indicate that the rate of propagation of IGA through thermally treated Inconel 600 was 2 or 3 times less than the rate of propagation through the mill annealed tube material. A larger reduction applies to the rate of propagation of SCC.[64]

---

63 Sleeving Report, Chapter 6.0; SER at 20, 23.

64 Fletcher, ff. Tr. 1422, at 6-7; Murphy, ff. Tr. 1828, at 2; Tr. 1483-88 (Fletcher).

2. Hydrostatic Testing

Previous to the time that sleeved tubes are placed in service,[65] and periodically thereafter,[66] applicant will perform hydrostatic tests to locate leaks in tubes. The tests involve pressure differentials substantially in excess of normal operating pressure differentials. The pressure differentials approximate those that would be expected to occur during postulated main steam line breaks or loss of coolant accident (LOCA) events.[67]

3. Continuous Leak Monitoring

Since primary water contains small amounts of radioactivity that may be detected if it migrates to the non-radioactive secondary side of the steam generator, applicant continuously monitors the secondary system condenser air ejector and steam generator blowdown for radioactivity. The presence of radioactivity in these locations would indicate a leak in the steam generator tubes or sleeves. Even very small leaks in tube sleeves can be detected through this monitoring process.[68]

---

[65] See Safety Evaluation by the Office of Nuclear Reactor Regulation relating to Full Scale Steam Generator Tube Sleeving at Point Beach Nuclear Plant Units 1 and 2, Docket Nos. 50-266 and 50-301, July 8, 1982 (SER), at § 6.0, p. 34.

[66] Murphy, ff. Tr. 1828 at 2, 10; Fletcher, ff. Tr. 1422 at 5.

[67] Murphy, ff. Tr. 1828 at 2, 10; Fletcher, ff. Tr. 1422 at 5; SER at 34-35 (approving hydrostatic test plans for mechanically sleeved joints and questioning the adequacy of differential pressures for testing applicant's abandoned plan for an alternate type of brazed upper joint).

[68] Fletcher, ff. Tr. 1422 at 5-6; Murphy, ff. Tr. 1828 at 2, 10.

The NRC has established limits on overall leakage from tubes in a steam generator. If those limits are exceeded, either by leaks through existing tubes or through sleeves, applicant will be required to shut down the reactor for repair. Although some leakage is permitted in recognition of the difficulty of installing entirely leak-tight tubes, leak limits are established in order to assure that the unit would be shut down before the integrity of the leaking tube or tubes could become sufficiently impaired to risk a rupture either under normal operating conditions or postulated accident conditions.[69]

Leak limits are so rigorous that even if the entire leakage occurred through one sleeve, the maximum through-wall crack length that could exist without exceeding the limits for leakage (500 gpd or 0.3 gpm per steam generator) would be about 0.4" at normal operating pressures. Even should a steam line break accident occur at a time that a flaw of that dimension existed, analysis indicates that the sleeve could withstand the increased pressure differential without bursting.[70]

4. Leak-Before-Break Characteristic of Sleeves

Another safety factor is that steam generator tubes and sleeves are made of a special material, Inconel 600, selected because of its high ductility and toughness, two characteristics which in combination constitute fracture resistance. In this material, a crack (SCC or IGA) that began to form on the tube or sleeve's outer wall probably would

---

69 Id.

70 Fletcher, ff. Tr. 1422, at 8.

cause a small, detectable leak before it became susceptible to a rupture either during accident or normal operating conditions.[71]

Laboratory and operating experience confirm the validity of the leak-before-break concept. Degraded tubes normally do not suffer large breaks; they penetrate locally, suffering only minor in leakage that is readily detectable through continuous leak monitoring. Almost all leakage events in Westinghouse steam generators were of this kind.[72]

Considering all operating reactors, there are hundreds of steam generators, containing thousands of tubes. In all the years of operation of these tubes, there have been approximately 200 leaks reported to the NRC, and only four of these have involved large leak rates. None of the four occurrences resulted in any unacceptable offsite radiological consequences or any damage to the reactor core. All resulted from unusual circumstances that do not invalidate the leak-before-break characteristic of steam generator tubes.

Important exceptions to the leak-before-break concept have emerged: that hoop stresses (caused by denting at the uppermost tube support plate), mechanical damage from loose parts,[73] and substantial thinning[74] may cause a rapid failure. However, there is no significant

---

[71] Fletcher, ff. Tr. 1422 at 7.

[72] Fletcher, ff. Tr. 1422, at 8.

[73] Murphy, ff. Tr. 1828, at 10; Tr. 1774-78 (Fletcher); <u>see also</u> Marsh, ff. Tr. 1822, at 3.

[74] Tr. 1774-81 (Fletcher).

denting present at Point Beach.[75] Applicant is aware of the loose parts problem and has begun monitoring for their presence.[76] Furthermore, eddy current testing can reliably detect seriously thinned tubes, all of which have been removed from service at Point Beach.[77] The basic concept, that tubes and sleeves will respond to corrosion by leaking before they break, is still applicable to the sleeving repair at Point Beach.

In addition to operating experience, conservative analyses substantiate the leak before break concept. The maximum primary-to-secondary pressure differential occurs following a postulated feedline break or steam line break accident, which reduces the secondary side pressure to zero. Analysis of this accident condition for the sleeve indicates that even if there is uniform thinning completely around the circumference, a sleeve can degrade to 38% of its nominal wall thickness and still resist rupture.[78] This corresponds to 62% degradation, or over 50% more

---

75 Licensee's response to Questions in Memorandum and Order, dated April 7, 1982. Although there has been some denting in Unit 2, it has not progressed significantly and does not constitute significant tube plate support deformation. Furthermore, these phenomena are not related to sleeving. Id. at 1-2.

76 Letter to the Atomic Safety and Licensing Board from Bruce Churchill, November 9, 1982.

77 Tr. 1774-81 (Fletcher). (Because phosphate chemistry is no longer in use at Point Beach, Mr. Fletcher does not expect new instances of thinning to occur.)

78 Sleeving Report at 6.120-6.121.

degradation than the 40% degradation whose detection -- at any one spot on the tubewall -- causes the NRC to require plugging of the tube.[79]

To further confirm the analyses, there have been laboratory tests. These "burst tests" have been performed on portions of tubes removed from Point Beach and suffering from IGA of about 40% to 60%. This testing required differential pressures in excess of 5000 psi to cause bursting of the degraded tubes. This indicates substantial additional margin over the conservatively estimated pressures resulting from postulated accidents.[80]

Over all, we are confident that the leak-before-burst concept, under normal operating conditions and postulated accident conditions, is applicable to the Point Beach sleeving amendment.

5. Conservative Criteria for Eddy Current Testing

At Point Beach, hydrostatic testing and eddy current testing programs reduce the risk that serious degradation of tube or sleeve walls may occur without detection. Both tubes and sleeves in which eddy current testing indicates 40% or more degradation must be removed from service.[81] Even though tubes and sleeves with small leaks are not subject to rupture, these testing programs successfully identify

---

79 Fletcher, ff. Tr. 1422, at 9; Murphy, ff. Tr. 1828, at 3-4.

80 Fletcher, ff. Tr. 1422, at 6-7; Murphy, ff. Tr. 1828, at 2; Tr. 1483-88 (Fletcher).

81 SER at 21.

partially degraded tubes, and those tubes are removed from service as an added precaution.

As we have just pointed out in the previous subsection of this opinion, eddy current test indications of 40% degradation cause tubes and sleeves to be removed from service. This represents more than a 50% safety margin, even were the degradation to be uniform for the entire outer diameter of the tested tubes.

We are convinced that eddy current testing, used in this conservative manner, contributes to the overall safety of the sleeved tubes.

6. Possible Leak Constraint from the Tube or Tubesheet

Most of the sleeved portion of the tubes lies within the tubesheet. In that area, which is the area in which IGA has been found when tube samples have been removed from the steam generator, the tube is tightly constrained by the tubesheet, minimizing any potential for rupture.[82] If rupture of the sleeve were nevertheless assumed to occur within the tubesheet as a result of IGA or SCC, the leak path would be obstructed by the narrow tube-to-tubesheet crevice, and the leak rate would be significantly reduced compared to the rate postulated to occur above the tubesheet from a ruptured tube.[83]

Sleeving would provide an additional barrier against leakage. Even if the sleeve begins to rupture, the event may be terminated or severely limited if it occurs in an area of the original tube which has

---

[82] Murphy, ff. Tr. 1283 at 6.

[83] Fletcher, ff. Tr. 1422, at 10; Murphy, ff. Tr. 1828, at 6.

sufficient remaining strength to resist rupturing at the corresponding point on the tubewall. If the tube does remain intact at that point, then it may prove an effective barrier to any leakage at all to the secondary side. In the alternative, leakage may occur into the sleeve-tube annulus and thence, through a hole in the sleeve, to the secondary side. However, such a leak undoubtedly would occur at a far slower pace than a fishmouth rupture or double-ended break in a single tube, not supported by a sleeve.[84] Even if these benefits of the sleeving configuration are not realized, there is no reason to believe that a rupture of a sleeve would be worse than the rupture of an unsleeved tube.[85]

7. Less Corrosive Environment in the Annulus

The rate of corrosion in tubes or sleeves depends on the environment to which they are exposed. The outer diameter of the sleeve will not be exposed to the secondary side environment unless degradation in the original tube propagates through-wall and the original tube's grain boundaries separate enough to admit solution from the non-pressurized secondary side into the annulus.[86] This would require substantially more degradation of the tube than would occur before it was removed from service because of fears that it could not withstand operating pressures

---

84 Marsh, ff. Tr. 1822, at 3-4; Murphy, ff. Tr. 1828, at 4.

85 Id.

86 Fletcher, ff. Tr. 1422, at 6.

or accident conditions. Hence, there ordinarily will be a substantial delay before the sleeve is exposed to a corrosive environment.

Should a corrosive environment occur in the annulus, the leak into the annulus would probably occur in the tubesheet area, where sludge is deposited. Thence, the corrosive material would travel to the bottom of the annulus, within the tubesheet crevice. In that location, it is possible that a corrosive environment could develop, but there is no reason to believe that the rate of corrosion would be any worse than what already is found in the tubesheet crevice. Consequently, the sleeves would never be exposed to a more corrosive environment than are tubes. Also, the location of the corrosion--at the bottom of the annulus--only creates a risk of a constrained leak, rather than a guillotine or fishmouth rupture.[87]

We have discussed, above, the testimony of Mr. Fletcher concerning the properties of the annulus and the reason for believing that the fluid turnover rate and sedimentation rate would be low in that area.

8. Conclusion

The uncontradicted evidence shows that sleeving enhances safety, both from the point of view of increased integrity of the primary

[87] Tr. 1767-70, 1766-73 (Fletcher); Tr. 1851-52 (McCracken); Tr. 1853 (Murphy). The implications of a constrained leak are discussed in subsection 6, supra.

pressure boundary and decreased consequences of a breach in the pressure boundary.[88] Sleeving will provide lower probabilities of the occurrence of the three events -- abnormal leakage, rapidly propagating failure, and gross rupture -- which are required to be minimized by General Design Criterion 14.[89] We therefore conclude that there is no serious safety or environmental issue of which we are aware that requires us to undertake our own further inquiry.

ORDER

For all the foregoing reasons and based on consideration of the entire record in this matter, it is this 4th day of February 1983

ORDERED:

1. The sole remaining genuine issue of fact in this proceeding, concerning the adequacy of eddy current testing of sleeved steam generator tubes, is dismissed.

2. We authorize the Director of Nuclear Reactor Regulation to issue a license amendment to Wisconsin Electric Power Company, concerning the repair of steam generator tubes at its Point Beach nuclear plant by sleeving, subject to understandings of record, that:

a. Steam generator tubes that have been previously subject

---

[88] We examined this question with especially great care because Mr. Marsh's testimony indicates that there may be a substantial risk from the rupture of only one or two steam generator tubes. Marsh, ff. Tr. 1822 at 5; Tr. 1839-41.

[89] Fletcher, ff. Tr. 1422, at 12.

to explosive plugging, shall not be sleeved;

b. Brazed joints shall not be employed;

c. Should eddy current testing indicate 40 percent or more degradation from the nominal tube wall thickness of a sleeve, the sleeved steam generator tube shall be plugged; and

d. Leak limits previously imposed on the repaired steam generators shall continue to apply.

3. Pursuant to 10 CFR § 2.760(a) this is an initial decision that will constitute final action of the Commission forty-five (45) days from the date of issuance unless exceptions are taken pursuant to § 2.762 or the Commission directs that the record be certified to it.

4. Exceptions to this decision or designated portions thereof may be filed with the Commission, in the form required by § 2.762(a), within ten (10) days after service of this decision.

5. To pursue an appeal, briefs in support of a party's objection also must be filed, within thirty (30) days after filing the exceptions (or forty days in the case of the staff of the Nuclear Regulatory Commission). The brief must comply with the requirements of § 2.762.

6. Within thirty (30) days of the service of the brief of the appellant (40 days for the staff), parties may file opposing or supporting briefs that comply with the requirements of § 2.762.

7. Filings that do not comply with the rules governing appeals may be stricken.

FOR THE
ATOMIC SAFETY AND LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Hugh C. Paxton
ADMINISTRATIVE JUDGE

Jerry R. Kline
ADMINISTRATIVE JUDGE

APPENDIX A

PUBLISHED POINT BEACH BOARD ORDERS

Title	Date of Document	LBP No.
MEMORANDUM AND ORDER (Requesting Additional Information)	10-01-81	81-39
MEMORANDUM AND ORDER (Setting Agenda For October 9 Conference Call)	10-07-81	81-43
MEMORANDUM AND ORDER (Concerning Further Board Questions)	10-13-81	81-44
MEMORANDUM AND ORDER (Concerning The Admission Of A Party And Its Contentions)	10-13-81	81-45
MEMORANDUM AND ORDER (Setting Agenda And Rules For October 29-30 Hearing)	10-15-81	81-46
MEMORANDUM AND ORDER (Authorizing Issuance Of A License Amendment Permitting Return To Power With Up To Six Degraded Tubes Sleeved Rather Than Plugged)	11-05-81	81-55
MEMORANDUM AND ORDER (Concerning Preliminary Confidentiality Issues)	12-21-81	81-62
SUPPLEMENTARY ORDER (Concerning Issuance Of A Protective Order)	01-07-82	82-2
MEMORANDUM AND ORDER (Concerning Reconsideration Of Confidentiality Issues)	01-28-82	82-5A

MEMORANDUM AND ORDER (Concerning The Burden Of Going Forward On Confidentiality Issues)	02-02-82	82-6
MEMORANDUM AND ORDER (Concerning A Motion To Compel And Other Matters)	02-19-82	82-10
MEMORANDUM AND ORDER (Concerning a Motion To Certify A *Sua Sponte* Question)	02-26-82	82-12
MEMORANDUM AND ORDER (Concerning A Motion To Reconsider)	03-19-82	82-19A
MEMORANDUM AND ORDER (Concerning Reconsideration Of A Motion To Certify A *Sua Sponte* Question)	03-31-82	82-24A
MEMORANDUM AND ORDER (Concerning A Motion To Compel)	04-22-82	82-33
MEMORANDUM AND ORDER (Concerning A Motion To Release To The Public Certain Safety Information Which Is Part Of The Record In This Case But Is Proprietary To Westinghouse Electric Corporation)	05-26-82	82-42
MEMORANDUM AND ORDER (Concerning Summary Disposition Issues)	10-01-82	82-88

Appendix B
Qualifications of Witnesses

Applicant's Witnesses

W.D. Fletcher, Manager of Steam Generator Development and Performance Engineering in the Nuclear Technology Division of the Westinghouse Electric Corporation. He has a Masters degree in Chemistry from Fordham University, 1960. Since 1970 he has been directly involved in development and design activities related to Westinghouse steam generators. He is credited with a variety of professional publications, including publications about Westinghouse steam generators, primary coolant chemistry in PWR's and corrosion of stainless steel.

Clyde J. Denton, a participant in the group that originated eddy current testing of steam generators and presently general manager of Zetec, Inc. He has an A.A.S. from the Milwaukee School of Engineering and has been doing eddy current testing since 1956.

Edward O. McKee, a technician with 11 years' experience in interpreting eddy current data. He has evaluated all ECT data for both Point Beach units.

Staff's Witnesses

Emmett L. Murphy, Senior Systems Engineer in the staff's Operating Reactors Assessment Branch. He has a Master of Science Degree in Civil Engineering and a Bachelor of Science Degree in Aerospace Engineering, both from the University of Maryland. He has worked for nine years in the nuclear field, including six years as structural engineer at the Bettis Atomic Power Laboratory of Westinghouse Corporation. Since July 1979 he has been working for the staff almost exclusively on safety reviews of steam generators that have experienced significant tube degradation.

Ledyard B. Marsh, Section Leader of staff's Reactor Systems Branch. He has a Masters of Science in Nuclear Engineering from the University of Washington, was an officer in the Navy Nuclear Power Program from 1970 to 1974, and joined the Reactor Systems Branch in 1976.

Timothy G. Colburn, staff's Project Manager for the Point Beach reactors. He has a B.S. in mechanical engineering from Notre Dame, worked in the Navy's nuclear power program and was employed by Potomac Electric Power Company.

Conrad E. McCracken, Section Leader of the staff's Chemical Technology Section of the Engineering Branch. He is a registered Professional Corrosion Engineer who was qualified in submarines for all nuclear duties by the United States Navy and who served as Manager of Chemistry Development for Combustion Engineering Corporation from 1966 to 1981, when he jointed the staff as a senior chemical engineer.

Note: Wisconsin's Environmental Decade did not call any witnesses.

APPENDIX C

Comment on Limited Appearance Statements

In preparing this decision, we remember the people who addressed us when we sat in a Limited Appearance session in Two Rivers Wisconsin on November 17, 1982. Although there are many people living near Point Beach who are pleased with the use of nuclear reactors to generate electricity,[90] the people who addressed us were thoughtful people with serious doubts. One of the speakers, Mr. Edward Klessig, said what many had on their minds:

> We pride ourselves on being practical farmers. We service most of our own equipment. The proposed sleeving repair process reminds us of fixing a sophisticated hay bailer or combine with a piece of bailing wire.
>
> As farmers and food producers we love the land. We don't want to risk contaminating the precious soil and the food chain with radioactive isotopes, at best, or total disaster at worst.[91]

We are aware of these citizen concerns and of the trust that is placed in us to resolve the matter before us. We are particularly aware that a license amendment dealing with "tube sleeving" does superficially

---

[90] The Town Board of the Town of Two Creeks unanimously supports the "economic and efficient way of producing electricity" at Point Beach and approves of the proposed sleeving process. Letter to Mr. Peter Bloch (November 29, 1982).

[91] Tr. 10009.

resemble a patchwork repair. Consequently, we have been especially attentive to our record, which contains numerous tests and analyses that have been relevant to our deliberations either at this or at an earlier stage of the proceeding.

We hope that if Mr. Klessig and his fellow citizens should read this memorandum that they will be assured that the steam generator repair has been engineered with great care. Even should they disagree with our conclusion that none of Decade's contentions is valid and that there is no serious safety or environmental issue for us to raise ourselves, we hope they will realize that our decision to approve the pending license amendment has not been lightly taken.

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. 71
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 amended by letter dated March 9, 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 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. 71 , 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: April 4, 1983

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-27

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 amended by letter dated March 9, 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 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-27 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76, 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: April 4, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
	15.1-6
15.3.1-9	15.3.1-9
15.3.1-10	15.3.1-10
	15.3.1-10a
	Figure 15.3.1-5
Table 15.4.1-2 (pg. 1 of 2)	Table 15.4.1-2 (pg 1 of 3)
(pg. 2 of 2)	(pg 2 of 3)
	(pg 3 of 3)
15.4.2-1c	15.4.2-1c
Table 15.4.2-1	Table 15.4.2-1
15.6.9-10	15.6.9-10

p. Dose Equivalent I-131

Dose Equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

q. $\overline{E}$ - Average Disintegration Energy

$\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

C. MAXIMUM COOLANT ACTIVITY

Specification:

The specific activity of the reactor coolant shall be limited to:

1. Less than or equal to 1.0 microcurie per gram Dose Equivalent I-131.

   a. If the specific activity of the reactor coolant is greater than 1.0 microcuries per gram Dose Equivalent I-131 but within the allowable limit (below and to the left of the line) shown on Figure 15.3.1-5, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. Reactor Coolant Sampling shall be in accordance with Table 15.4.1-2. A Special Report shall be prepared in accordance with specification 15.6.9.3.F if cumulative operating time above exceeds 500 hours in any consecutive 6-month period.

   b. If the specific activity of the reactor coolant is greater than 1.0 microcuries per gram Dose Equivalent I-131 for more than 48 hours during one continuous time interval or exceeds the allowable limit (above and to the right of the line) shown on Figure 15.3.1-5, the reactor will be shut down and the average reactor coolant temperature will be less than 500°F within 6 hours.

2. Less than or equal to $100/\bar{E}$ microcuries per gram.

   a. If the specific activity of the reactor coolant is greater than $100/\bar{E}$ microcuries per gram, the reactor will be shut down and the average reactor coolant temperature will be less than 500°F within 6 hours. Reactor Coolant Sampling shall be in accordance with Table 15.4.1-2.

3. Reportable Occurrences required by specification 15.6.9.2.3.2 for the above conditions shall contain the results of the specific activity analyses together with the following information:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

b. Fuel burnup by core region,

c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

d. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

e. The time duration when the specific activity of the primary coolant exceeded 1.0 microcuries per gram DOSE EQUIVALENT I-131.

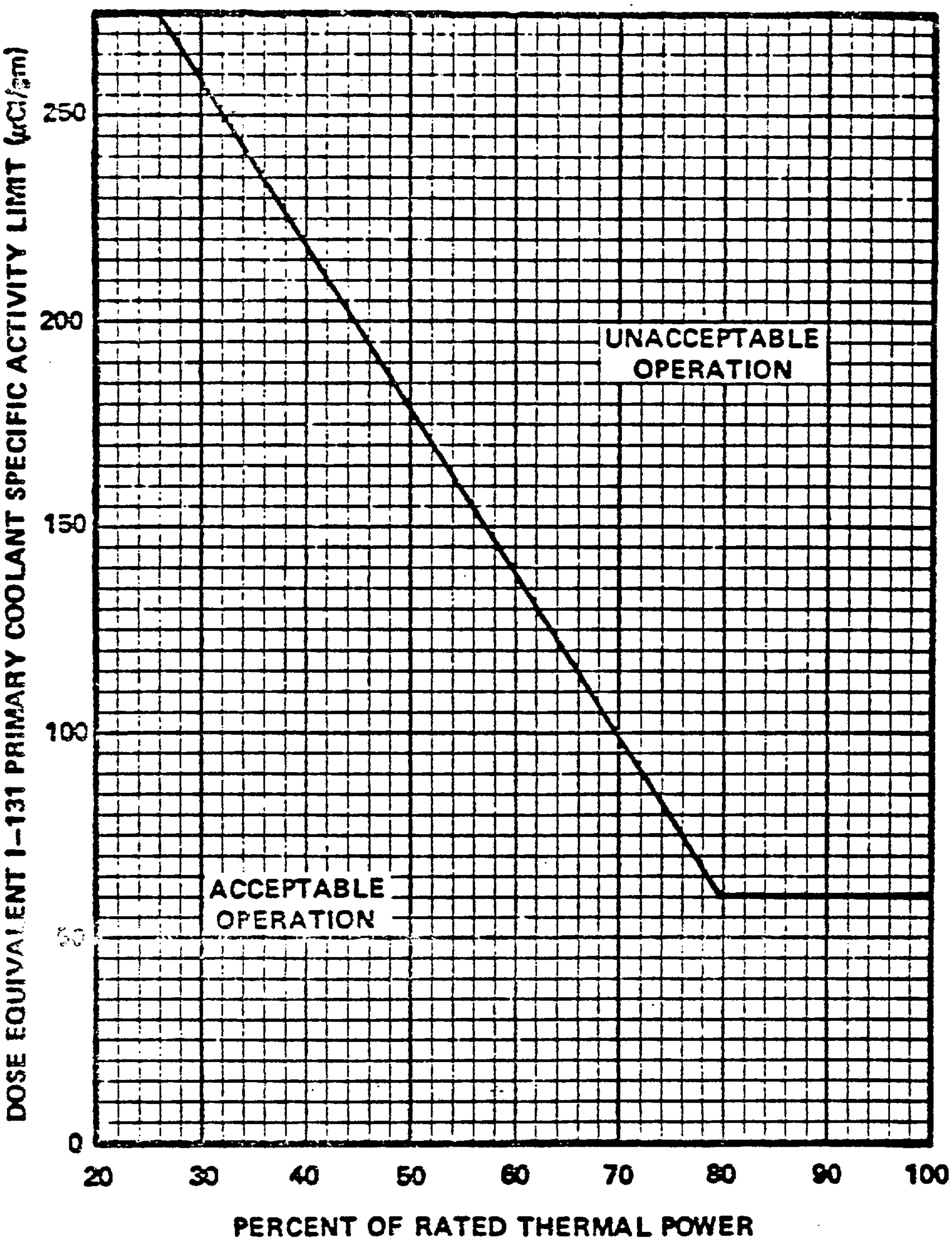
Basis:

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative for Point Beach Nuclear Plant.

Continued power operation for limited time periods with the reactor coolant's specific activity greater than 1.0 microcurie/gram Dose Equivalent I-131, but within the allowable limit shown on Figure 15.3.1-5, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation with specific activity levels exceeding 1.0 microcurie/gram Dose Equivalent I-131 but within the limits shown on Figure 15.3.1-5 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to less than 500°F normally prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

FIGURE 15.3.1-5



DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

Unit 1 - Amendment No. 71
Unit 2 - Amendment No. 76

TABLE 15.4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	Test	Frequency
1. Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week (7)
	Tritium activity	Monthly
	Radiochemical $\overline{E}$ Determination	Semiannually (2)(11)
	Isotopic Analysis for Dose Equivalent I-131 Concentration	Every two weeks (1)
	Isotopic Analysis for Iodine including I-131, I-133, and I-135	a) Once per 4 hours whenever the specific activity exceeds 1.0μCi/gram Dose Equivalent I-131 or 100/$\overline{E}$ μCi/gram. (6) b) One sample between 2 and 6 hours following a thermal power change exceeding 15% of rated power in a one-hour period.
	Chloride Concentration	5/week (8)
	Diss. Oxygen Conc.	5/week (6)
	Fluoride Conc.	Weekly
2. Reactor Coolant Boron	Boron Concentration	Twice/week
3. Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly (6)
4. Boric Acid Tanks	Boron Concentration	Twice/week
5. Spray Additive Tank	NaOH Concentration	Monthly
6. Accumulator	Boron Concentration	Monthly
7. Spent Fuel Pit	Boron Concentration	Monthly
8. Secondary Coolant	Gross Beta-gamma activity or gamma isotopic analysis	Weekly (6)
	Iodine Concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.2 μCi/cc (6)

		Test	Frequency
9.	Control Rods	Rod drop times of all full length rods (3)	Each refueling or after maintenance that could affect proper functioning (4)
10.	Control Rod	Partial movement of all rods	Every 2 weeks (6)
11.	Pressurizer Safety Valves	Set point	Each refueling shutdown
12.	Main Steam Safety Valves	Set point	Each refueling shutdown
13.	Containment Isolation Trip	Functioning	Each refueling shutdown
14.	Refueling System Interlocks	Functioning	Each refueling shutdown
15.	Service Water System	Functioning	Each refueling shutdown
16.	Primary System Leakage	Evaluate	Monthly (6)
17.	Diesel Fuel Supply	Fuel inventory	Daily
18.	Turbine Stop and Governor Valves	Functioning	Monthly (6)(10)
19.	Low Pressure Turbine Rotor Inspection (5)	Visual and magnetic particle or liquid penetrant	Every five years
20.	Boric Acid System	Storage Tank temperature	Daily
21.	Boric Acid System	Visual observation of piping temperatures (all $\geq$ 145°F)	Daily
22.	Boric Acid Piping Heat Tracing	Electrical circuit operability	Monthly
23.	PORV Block Valves	Complete Valve Cycle	Quarterly (6)
24.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Yearly
25.	Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly (9)

(1) Required only during periods of power operation.
(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates $\geq$ 10 μc/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10 μc/cc.

Table 15.4.1-2 (Continued)

(3) Drop tests shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot conditions, but cold drop tests need not be timed.

(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.

(5) As accessible without disassembly of rotor.

(6) Not required during periods of refueling shutdown.

(7) At least once per week during periods of refueling shutdown.

(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.

(9) Not required during periods of cold or refueling shutdown.

(10) During end of cycle period of operation when boron concentration is less than 100 ppm, this test may be waived due to operational limitations.

(11) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.

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 or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

6. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving* prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged.

7. Reports

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in the Annual Results and Data 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.

(c) Reports shall include:

1. Number and extent of tubes inspected.
2. Location and percent of all thickness penetration for each indication.
3. Identification of tubes plugged or repaired.

(d) Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection shall provide the information required by Specification 15.4.2.A.7(b) 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 Safety Class Components Other than Steam Generator Tubes

1. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) modified by Section 50.55a(b), except where specific written relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

---

*Brazed joints shall not be employed. Tubes previously subject to explosive plugging shall not be sleeved.

TABLE 15.4.2-1

STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.) S=3(N/n)% where: N is the number of steam generators in the plant = 2 n is the number of steam generators inspected during an examination	C-1	Acceptable for continued service	N/A	N/A	N/A	N/A
	C-2	Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
			C-2	Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
					C-2	Plug or repair tubes exceeding plugging limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Report results to NRC within 24 hours in accordance with TS 15.6.5.2.A.3	C-1 in other S.G.	Acceptable for continued service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Report to NRC within 24 hours in accordance with TS 15.6.5.2.A.3.	N/A	N/A

Unit 1 - Amendment No.
Unit 2 - Amendment No.

(1) The number and types of samples taken and the measurements made on the samples; e.g., gross beta gamma scan, etc.

(2) Any changes made in sample types or locations during the reporting period, and criteria for these changes.

b. A summary of survey results during the reporting period.

4. <u>Leak Testing of Source</u>

Results of required leak tests performed on seal sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. <u>Poison Assembly Removal From Spent Fuel Storage Racks</u>

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

E. <u>Overpressure Mitigating System Operation</u>

In the event the overpressure mitigating system is operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

F. <u>Dose Equivalent I-131</u>

With total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie per gram Dose Equivalent I-131 exceeding 500 hours in any consecutive 6-month period, submit a report within 30 days indicating the number of hours above this limit.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has, pursuant to the Initial Decision of its Atomic Safety and Licensing Board (ASLB) dated February 4, 1983, (ASLBP No. 81-464-05 LA) issued Amendment Nos. 71 and 76 to Facility Operating License Nos. DPR-24, and DPR-27 issued to Wisconsin Electric Power Company (the licensee), which revised Technical Specifications (TS) for operation of Point Beach Nuclear Plant Unit Nos. 1 and 2 (the facilities) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective as of the date of issuance.

The amendments to the TS allow repair of degraded steam generator tubes by sleeving which would otherwise be required to be plugged and removed from service; establish limits for primary coolant iodine concentration and surveillance frequency; and establish a plugging limit for sleeved tubes of 40% nominal sleeve wall thickness.

The Initial Decision is subject to review by an Atomic Safety and Licensing Appeal Board prior to its becoming final. Any decision or action taken by an Atomic Safety and Licensing Appeal Board in connection with the Initial Decision may be reviewed by the Commission.

The amendments comply 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 amendments.

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 (46FR 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 November 17 and 18, 1982 with limited appearances held in the town of Two Rivers, Wisconsin on the evening of November 17, 1982. The Board issued its Initial Decision on February 4, 1983 and ruled that the NRC staff was authorized to issue the amendments.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendments dated July 2, 1981 as amended March 9, 1983, (2) the Initial Decision of the Atomic Safety and Licensing Board dated February 4, 1983, (3) Amendment Nos. 71 and 76 to Facility Operating Licenses No. DPR-24 and DPR-27, and (4) the Commission's letter to the licensee dated April 4, 1983 . 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 4th day of April, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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