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NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

Wisconsin Electric Power Company
POINT BEACH NUCLEAR PLANT UNIT 1
DOCKET NO. 50-266
Operating License Amendment 2
(Steam Generator Replacement Proceeding)

DECADE'S CONTENTIONS CONCERNING STEAM GENERATOR REPLACEMENT

Pursuant to 10 C.F.R. §2.714(b), and the criteria set forth in Re Cleveland Electric Illuminating Company, 14 N.R.C. 175, 184(1981), Wisconsin's Environmental Decade, Inc.("Decade"), hereby supplements its Petition to Intervene and Petition for Hearing, dated August 10, 1982, by listing the contentions which, at this time, it seeks to have litigated.

The Decade makes this filing without waiving its previously stated objection to the scheduling order of the Atomic Safety and Licensing Board ("Board") that requires an impecunious party to make a detailed filing in this proceeding at the same time it is required to prepare direct testimony in a companion proceeding, Docket 50-266 OLA-1. In order to avert legal waiver of the right to participate in this proceeding, it has been necessary for us to involuntarily relinquish the opportunity to secure affirmative testimony in the companion proceedings, because both tasks could not be accomplished simultaneously. The objectionable nature of the scheduling order is compounded further by the fact that the

Licensee has made no showing, nor has it been required to make a showing as we requested, to demonstrate the need for such an expeditious schedule that has as its intended impact the fatal impairment of the interests of the opposing citizenry.

Although listed separately for ease of reference, the following contentions are all interrelated with each other and should be considered as such.

FIRST CONTENTION
Tube Failures Under LOCA Accident Conditions

Contention

Degradation of as few as one to ten steam generator tubes in either the existing or the proposed steam generators at Point Beach Nuclear Plant Unit 1 ("Point Beach") could induce essentially uncoolable conditions in the course of a loss-of-coolant-accident ("LOCA"), a condition which was not considered by the Nuclear Regulatory Commission ("Commission") with regard to the existing generators prior to licensing the facility, in the Final Safety Analysis Report or in any subsequent license amendment proceeding, nor which is addressed in the application for the proposed generators.

These factors from secondary-to-primary leakage through degraded steam generator tubes act to lower the threshold for admitting contentions such as to make a matter with a low probability justiciable, even if it might otherwise not be so, due to the large consequences from its occurrence. Further, inasmuch as these factors were not evaluated as part of the original operating license, it is necessary that they be evaluated in this proceeding to amend the operating license prior

to its being approved.

Basis

"The basis for our concern about the present course of actions being pursued by the task force * * * lies in the indeterminancy of the adequacy of the present code formulations. * * * [A] clear demonstration of coolability by wide margins is necessary to satisfy this uncertainties[sic] regarding the ECCS capability; that is, cooling by narrow margins would have to be regarded by him as an essentially uncoolable situation. * * * Some of the essential areas of uncertainty in predicting ECCS performance are reflooding and steam binding. * * * Of paramount concern in this area, however, is the possible effect of steam generator tube failures on the ECCS." REG ECCS Task Force, Memorandum to ECCS Task Force Members, dated June 16, 1972.

"[I]t was the consensus of the [American Physical Society] group that steam generator tube failure during a severe LOCA could occur frequently. Moreover, it appears that rupture of a few tubes (on the order of one to ten) dumping secondary steam into the depressurized primary side of th reactor system could exacerbate steam binding problems and induce essentially uncoolable conditions in the course of a LOCA * * *." Report to the American Physical Society by the Study Group on Light-Water Reactor Safety, 47 Review of Modern Physics(Summer 1975), at p. S85.

"Furthermore, serious weakening of these tubes from similar causes [of tube degradation] could, in the event of a loss-of-coolant-accident (LOCA), result in tube failures that would release the energy of the secondary system into the containment." Regulatory Guide 1.83(Rev. 1), at p. 1.

"If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECC preventing adequate cooling of the core. The critical number of tubes is relatively small." Office of Nuclear Reactor Regulation, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, NUREG-0410(1978), at p. C-29.

"The failure of a number of steam generator tubes as a result of the pressure transients during a loss of coolant accident could render the emergency core cooling system ineffective." Risk Assessment Review Group, Report to the U. S. Nuclear Regulatory Commission, NUREG/CR-0400(1978), at p. 48.

"Recent studies have shown that as few as ten tubes would need to have ruptured during a LOCA (assuming a leakage rate of 130 gal/min per ruptured tube) before the

cladding temperature would be significantly affected (i.e. peak cladding temperature (PCT) [greater than] 2200°F)." Evaluation of Steam Generator Tube Rupture Events, NUREG-0651(1978), at p. I-2("Event Evaluation Report").

"One area [of research] that has not been considered sufficiently using recent accident analysis codes is estimation of the consequences of a transient or some other failure that might lead in turn to the failure of a significant number of tubes. Such failures could lead to the degradation of ECCS function." Office of Reactor Safety Research Group, Report to the President's Nuclear Safety Oversight Committee(1981), at p. I-2.

"The consequences of multiple tube failure, excess of the design base, have not yet been rigorously studied. * * * In the event of a LOCA, the core reflood rate could be retarded by steam binding. * * * S[team] G[enerator] tube failures would create a secondary to primary leak path which aggravates the steam binding effect and could lead to ineffective reflooding of the core." Nuclear Reactor Research, Steam Generator Status Report(Feb. 1982), at p. 2 to 3("Status Report").

"At the times Point Beach Unit 1, Surry Unit 2, and Prairie Island Unit 1 were licensed, there were no specific analysis requirements for S[team] G[enerator] T[ube] rupture events. * * *

"* * *

"The staff does not require licensees to analyze loss-of-coolant accidents (LOCAs) concurrent with an SGT break, but does require all LOCA analyses to include the effects of the plugged tubes on reduced RCS flow." Event Evaluation Report, at p. 1-2.

"The purpose of this section is to evaluate the impact, if any, of the repaired [sic] steam generators on the accident analysis transients for Point Beach Unit 1. Under the guidelines specified in 10 CFR 50.59 such an evaluation is required to verify that no unreviewed safety concerns or changes to the Technical Specifications occur. This section provides a qualitative discussion of the effect on the accident analysis of steam generator parameter changes from steam generator repair. Conclusions are made concerning the applicability of the original F[inal] S[afety] A[nalyses] R[eport] to the repaired [sic] unit. Consistent with the requirements of 10 CFR 50.59, licensing regulations and guidelines of the original licensing of the Point Beach Unit are assumed to apply, and only changes in the safety analysis due to equipment changes are considered." Wisconsin Electric Power Company, Steam Generator Report for Point Beach Nuclear Plant Unit 1, Docket 50-266, dated August 1982 ("Steam Generator Replacement Report"), at p. 5-1.

SECOND LITIGABLE ISSUE
Tube Failures Under Normal Operation Conditions

Contention

Rupture of steam generator tubes during normal operation may release radiation to the environment from the plant's secondary side in excess of maximum permissible doses to the extent that:

(a) Iodine. The iodine levels in the primary coolant exceed presently effective Westinghouse Standard Technical Specifications for reactor coolant iodine activity.

(b) Unconsidered Leakage. The primary-to-secondary leakage is greater than bounded in the Final Safety Evaluation Report for Point Beach ("FSAR") or in the Steam Generator Replacement Report due to such things as multiple tube failures or single tube ruptures greater than assumed in the design basis analysis.

(c) Safety Valve. The secondary side safety valve set point is exceeded and does not properly reseat for an extended period.

(d) Main Steam Line Break. Primary leakage through a ruptured tube overfills the steam generator and floods the main steam line with water that causes a main line steam break.

(e) Condensor. The condenser is removed from service during a tube rupture accident due to mechanical or economic reasons and the iodine partitioning function is lost.

These factors from primary-to-secondary leakage though degraded steam generator tubes act to lower the threshold for

admitting contentions such as to make a matter with a low probability justicible, even if it might otherwise not be so, due to the large consequences from its occurrence. Further, inasmuch as these factors were not, in large part, evaluated as part of the original operating license, it is necessary that they be evaluated in this proceeding to amend the operating license prior to its being approved.

<u>Basis</u>

Iodine

> "A steam generator tube rupture occuring while the reactor coolant iodine concentration is above the level allowed by the Westinghouse Standard Technical Specifications could cause offsite doses exceeding 10 CFR Part 100 guidelines." Safety Evaluation Report Relating to Full Scale Steam Generator Tube Sleeving at Point Beach Nuclear Plants Unit 1 and 2, Dockets 50-266 and 50-301 OLA-2, undated but presumed issued July 8, 1982 ("Sleeving SER"), at p. 43.

Unconsidered Leakage

> "The consequences of multiple tube failues, in excess of the design base, have not yet been rigorously studied. Rapid degradation between inspections of a large number of tubes could create the potential for multiple tube failures in the event of a plant transient or failure of a single tube and the accompanying jet impingement and the tube whip could cause failure of additional tubes. Furthermore, the potential for complicating circumstances involving multiple equipment failures such as the stuck open PORV during the Ginna incident and possible steam bubble formation in the primary system have not been evaluated. Another concern is ruptures in multiple S[team] G[enerators]. In this event, unless the plant can be rapidly depressurized and brought onto Residual Heat Removal, there is the potential to continuously lose emergency core cooling water outside of containment." Status Report, at p. 2.

> "The FDSA [for Ginna Nuclear Plant] predicted that with a double ended guillotine break of a single steam generator tube, the primary to secondary leak rate would be about 843 gpm. The initial leak rate at Ginna was calculated to be about 760 gpm, even though the break was not a double ended guillotine break.
>
> "During the January 25, 1982 incident at Ginna, the total amount of primary-to-secondary leakage and the total

amount of water and steam released to the environment were larger than would normally be predicted, because of valve malfunctions and operator actions (see Chapters 3, 4, and 5 of NUREG-0909). A comparison with a previous safety evaluation report input on the radiological consequences of a steam generator tube rupture accident (SGTR) (ref 8.3) shows that the potential exists for doses exceeding Part 100 Guidelines from a design-basis SGTR accident. * * * Although a more serious event was avoided and the radioactivity releases were not excessive, the staff concluded that additional measures must be taken to prevent potential accidents in the future from having similarly large leakages and releases that could cause more severe radiological consequences." Ginna SER, at p. 3-7.

Safety Valve

"In the actual event [on January 25, 1982 at Ginna Nuclear Plant], the damaged B steam generator safety valve lifted five times. After the last operation, the safety valve apparently failed to fully close. This failure was not explicitly considered in the FDSA, nor was its effect of continued primary-to-secondary leakage." Ginna SER, at p. 3-7.

"However, a slow RCS pressure reduction may, without careful operator attention, result in opening of the damaged SG ADV or safety valve(SV). The resulting offsite doses could be significantly greater than those experienced [in prior tube rupture events]." Event Evaluation Report, at p. 2-4.

Main Steam Line Break

"The FDSA assumed reducing safety injection flow to drop the RCS pressure to below 1100 psi at either one hour or four and one-half hours. In the actual event, plant pressure fluctuated above 1100 psia until two hours and 10 minutes after the tube rupture when it was bought [sic] to 1100 psig. This resulted in lifting the damaged steam generator safety valves a number of times. Even after RCS pressure was bought [sic] to below 1100 psia, leakage continued into the steam generator since the damaged steam generator pressure also dropped, probably the result of the leaking steam generator safety valve. Leakage into B S[team] G[enerator], resulted in excessive level and concern regarding the steam line ability to withstand the extra weight of being filled with water.

"Since the FDSA assumed time for primary and secondary pressure equilization did not result in S[team] G[enerator] overfill, and the FDSA did not analyze the resulting system (as well a offsite consequences - Section 8.1.3) effects, the FDSA does not bound the actual event in this respect." Ginna SER, at p. 3-6.

Condenser

"The staff's design-basis S[team] G[enerator] T[ube] R[upture] analyses presently include the assumption that the condenser is unavailable, since there are a number of reasons it may be unavailable after an accident; loss of offsite power; loss of instrument air; loss of service water resulting in loss of instrument air; the turbine bypass valves being out of service; and, as during the Ginna accident, a decision to remove the condenser from use. Without the condenser as a heat sink, there will be steam or steam/water release from the unaffected steam generator (always), and from the affected steam generator (for essentially all design basis accidents). In this event considerably more radioiodine is released and the dose consequences are dominated by the iodine releases. The [Ginna] licensee analyzed this type of accident, but the description of the analysis is too vague. The FDSA refers to 'all the primary iodine activity expected from 1% failed fuel,' without specifiying how many curies would be expected to be released to the coolant. The description says this release, via the atmospheric relief or code safety valves, 'could only occur if all offsite power to run the main condenser was lost and...' Since there were, in fact, considerable releases from the B steam generator code safety valve during the ginna accident, even though offsite power was available, the staff concludes that the licensee underestimated the probability of releases from paths other than the condenser. Secondary leakage is prolonged beyond the time typically assumed by the licensee and by the staff. Because this prolonged leakage both increases the secondary side fission product (mostly iodine) concentration, and decreases the iodine partitioning, it is important to consider the effects of continued leakage into the S[team] G[enerator] and overfilling with liquid rease out the S[afety] V[alve]." Ginna SER, at p. 3-9.

THIRD CONTENTION
Elimination of Crevice

Contention

The proposed steam generator will eliminate the tubesheet crevice where corrosive impurities have concentrated in the past by hyraulically expanding the new tubes to the full depth of the tubesheet holes. At the same time as the crevice is eliminated, however, this process will shift the roll stressed transition zone (between the expanded and unexpanded part of the tube) from near the bottom of the tubesheet hole to a point level with and

above the upper surface of the tubesheet. Compare Diagram C to Diagram B in Attachment 1. This will create four interrelated problems:

(a) Residual Stresses. The newly situated roll stressed transition zone will be subject to stress assisted cracking due to residual stresses from the hydraulic expansion process.

(b) Sludge Deposits. The zone will subjected to extensive corrosive attack, in addition to and compounded by stress assisted cracking, because it is located directly under deposits from impurities in the bulk secondary water that cannot be entirely eliminated in a pressurized steam generator of the existing or proposed design operating with an all volatile water chemistry treatment and also is in an deposition area subject to alternate wetting and drying.

(c) Detectability. It will be more difficult for eddy current testing to detect stress-assisted defects or corrosion in the transition zone than in the unexpanded portion of the sleeve.

(d) Unconstrained Leakage. Through-wall defects in the stressed and corroding transition zone of the proposed steam generators, unlike defects in the transition zone of the existing generators, will be unconstrained by the surrounding wall of the tubesheet, and the resulting secondary-to-primary in-leakage will lead to the safety concerns discussed in the First Contention and primary-to-secondary leakage, to the concerns in the Second Contention.

These problems with eliminating the crevice create a justiciable controversy as to whether the proposed steam generators, by their design, will suffer tube degradation, and do so in more ominous locations, and thereby fail to comply with applicable Commission regulations, 10 C.F.R. §50.40(a) ("the health and safety of the public will not be endangered"), and 10 C.F.R. Part 50 App. A Crit. 14 ("pressure boundary shall * * * have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture"), such as to mandate denial of an operating license amendment.

Basis

Introduction

> "Following insertion into the tubesheet hole, tack rolling, welding and gas leak testing, the tubes are hyraulically expanded to the full depth of the tubesheet holes. Full-depth closes eliminates the tube sheet crevice in which concentration of impurities has occured in the original steam generator." Steam Generator Replacement Report, at p. 2-8.

Residual Stresses

> "Westinghouse now, on new V[ertical] S[team] G[enerators], is applying a full roll to the [tubes at the bottom of the steam generator] to roll out the long crevices. When asked [by the utilities] if they had done research on the top of the rolled zone for possibilities of stress-assisted cracking, Westinghouse replied that they had not. The Westinghouse position is that the crevices are okay, but they are yielding to customer pressure to roll them out. That being the case, then Westinghouse should do research on the top of the roll-stressed area to keep from jumping from the frying pan in to the fire."
>
> "* * *
>
> "Westinghouse is rolling out the crevices, but is not concerned about the residual stress left in the Inconel 600 by rolling. Testing of rolled out specimens should be done under realistic environmental conditions." Ad Hoc Committee on Steam Generators, Final Report to the Edison Electric Institute Nuclear Plant Design and Operations Task Force on Pressurized Water Reactor Steam Generators, August 1, 1974, at Part VII, p. 2 ¶5 and p. 12 ¶32f.

"While the actual failure mode of the two leaking tubes cannot be known without removal and additional inspection, tube leaks identified on the inner row [of North Anna Power Station Unit 1] have been attributed to residual manufacturing stresses created during tube bending. * * *" Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Docket 50-338, dated December 10, 1979, at Attachment p. 4.

Sludge Deposits

"The use of 'zero solids treatment' is specifically not advised, owing to the lack of counteracting chemical treatment to absorb impurities introduced by condenser leakage.

"The alternate water chemistry is 'zero solids treatment in which hydrazine is used for oxygen control, and in some cases ammonia or a volatile amine is added for pH control. <u>Such treatment is not recommended.</u> It is considered risky because in the practical case the steam generator water chemistry will not be maintained truly free of solids. * * * All of the steam genrators have areas where the thermal and hyraulic conditions are such as to cause the 'drop out' of solids. Without the influence of proper phosphate adjustment of the boiler water chemistry, experience at Beznau I indicates that the resuling deposits can create an environment which is detremental to the tubing material." Westinghouse Electric Corporation, <u>Summary Paper on Beznau I Steam Generator Tube Leakage Problem</u>, May 10, 1972, at pp. 1 and 15.

Detectability

"The only place on the sleeves where cracks would be expected to have a circumferential orientation (if they were to occur) would be at the expansion transitions of the joints. Routine inspections with bobbin probes generally have not been capable of detecting circumferential flaws at similar joint transitions which already exist on the unsleeved tubes. Should such cracks occur, it will likely be necesary to employ a non-standard probe such as the pancake probe to detect these cracks. Circumferential cracks at expansion transitions have not generally been of concern since (1) such cracks typically involve only a small fraction of the tube circumference before resulting in a detectable leak and (2) even if complete severance of the tube occurred during accidents, the resulting leakage would be severely limited by the tubesheet crevice. For sleeves, the resulting leakage would be expected to be severely limited by the narrow sleeve to tube gap." Prefiled Testimony of Commission Staff Witness, Emmett L. Murphy, in the Full Scale Sleeving Proceeding, Docket 50-266 and 50-301, at pp. 9 to 10.

FOURTH CONTENTION
Balance of Plant

Contention

The replacement of the lower assemblies and moisture separators of the Point Beach steam generators will not serve to repair or substitute for other interrelated structural weaknesses in the balance of the plant, including the following:

(a) Condensers. The major source of corrodents in the steam generators in the past has been from leaks through failing condensor tubes. The condensers at Point Beach will not be replaced even though they do not meet present construction standards and remain a continuing source of tertiary-to-secondary in-leakage.

(b) Feedwater System. A new source of corrodents in the proposed steam generators may come from other plant components operated under the new water chemistry. The AVT water chemistry treatment that will be used may corrode pumps and piping that feed water to the steam generator of older plants such as Point Beach using copper based alloys, and cause degradation of the tubing from copper oxides. These components with copper alloys which modern standards discourage will not be replaced.

(c) Condensate Polishers. Because AVT does not absorb impurities, this water chemistry treatment is frequently coupled in new plants with a condensate polisher to remove the inevitable corrodents that will be part of the feedwater. No condensate polisher is proposed for inclusion within the operating license amendment.

These problems with corrosive impurities from other impaired plant components that will not be replaced or from the failure to install new components as a necessary adjunct to this license amendment, create a justiciable controversy as to whether the proposed steam generators, by their limited scope of repair, will continue to suffer tube degradation and thereby fail to comply with applicable Commission regulations, 10 C.F.R. §50.40(a) ("the health and safety of the public will not be endangered"), and 10 C.F.R. Part 50 App. A Crit. 14 ("pressure boundary shall * * * have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture"), such as to mandate denial of the operating license amendment.

Basis

Condensers

> "Design improvements [over admiralty based condensors at Point Beach] are recommended to bring condensers to modern construction standards. Westinghouse has proposed titanium tubes and titanium clad tubesheets [in condensers]." W. D. Fletcher(W), "Operating Experience with Westinghouse Steam Generators," 28 Nuclear Technology 357, March 1976, at p. 369.

> "Another area of improved design concentrates on the selection of more corrosion-resistant materials in the condenser because water leaks through the failed condenser tubing, when combined with air, can contaminate the condensate, feedwater, steam generator water, and steam. This contamination in turn degrades the structural integrity of the steam generator tubes, turbine, and other components in the cooling system. The utilities are eliminating the use of ammonia-sensitive alloys from the condensers and replacing them with more corrosion resistant alloy tubing. * * * The copper alloys are being replaced by materials such as titanium, AL 6X, or stainless steel (for freshwater service)." Steam Generator Tube Experience, NUREG-0886(February 1982)("Updated Experience Report"), at p. 37.

Feedwater System

"A major disadvantage of AVT is that the boiler water is unbuffered and subject to extensive and rapid pH excursions in the event of feedwater contamination. Further, excessive hydrazine can decompose producing ammonia; unacceptable amounts of ammonia in the condenser or feedwater train can corrode copper-base alloys and allow potentially-deleterious corrosion products to enter the steam generator." Testimony of Public Service Commission Witness, Dr. James R. Meyers, in Re Wisconsin Electric Power Company, Dockets 6630-ER-10 and 6630-UI-2, at Exhibit 68, p. 17.

"Until better control of the feedwater chemistry is established, we suggest that samples from the discharge of the high pressure feedwater heaters be monitored frequently for iron and copper. A review of steam generator crud analysis from Units 1 and 2 sampled February, 1973 shows the presence of Cu in steam generator crud at levels substantially higher than six other operating plants reviewed.

"As discussed in the previous chemistry review continued operation with elevated oxygen and ammonia concentrations in the feedwater presents a potentially deleterious chemical environment to this copper and ferritic system. Accelerated corrosion can occur under these conditions to the feedwater reheat system. These corrosion products would be transported to the steam generator thus contributing to the formation of sludge on the tube sheet." Letter from G. W. Hood(W) to C. A. Reed(WE), dated February 4, 1974, re Point Beach Chemistry Review, at p. 2.

Condensate Polishers

"The incorporation of full-flow condensate polishing (FFCP) into the feedwater cycle is viewed by many as a necessary adjunct to AVT." W. D. Fletcher(W), "Operating Experience with Westinghouse Steam Generators," 28 Nuclear Technology 357, March 1976, at p. 370.

FIFTH CONTENTION
All Volatile Treatment

Contention

The water chemistry treatment intended for use in the proposed steam generators is an all volatile treatment ("AVT"), instead of a congruent phosphate treatment that had originally been used in the existing generators. This creates four new problems:

(a) Solids Removal. AVT fails to perform the function of removing impurities from the bulk secondary water that had been performed by phosphates and which may otherwise lead to corrosive conditions in the steam generators.

(b) Detection. This problem of unprecipitated impurities with AVT is compounded in a pressurized steam generator such as Point Beach because detection is done largely in the bulk water and not in localized areas where corrodents concentrate and deposit.

(c) Feedwater Train. Excessive hydrazine with AVT can decompose producing ammonia which, in the feedwater train, can corrode copper-based alloys and allow corrosion products to enter the steam generator.

These problems with AVT create a justiciable controversy as to whether the proposed steam generators, in operation, will suffer from corroding tubes and thereby fail to comply with applicable Commission regulations, 10 C.F.R. §50.40(a) ("the health and safety of the public will not be endangered"), and 10 C.F.R. Part 50 App. A Crit. 14 ("pressure boundary shall * * * have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture"), such as to mandate denial of the operating license amendment.

Basis

Solids Removal

"The use of 'zero solids treatment' is specifically not advised, owing to the lack of counteracting chemical treatment to absorb impurities introduced by condenser leakage.

"The alternate water chemistry is 'zero solids treatment in which hydrazine is used for oxygen control, and

in some cases ammonia or a volatile amine is added for pH control. Such treatment is not recommended. It is considered risky because in the practical case the steam generator water chemistry will not be maintained truly free of solids. * * * All of the steam genrators have areas where the thermal and hyraulic conditions are such as to cause the 'drop out' of solids. Without the influence of proper phosphate adjustment of the boiler water chemistry, experience at Beznau I indicates that the resuling deposits can create an environment which is detremental to the tubing material." Westinghouse Electric Corporation, Summary Paper on Beznau I Steam Generator Tube Leakage Problem, May 10, 1972, at pp. 1 and 15. [Emphasis in original.]

"The primary advantage of AVT is that no dissolved solid additives are used (such as phosphates) which can concentrate in the steam generators to induce corrosion, such as phosphate wastage of Inconel-600 tubing. The disadvantage of AVTis that it provides no buffering capacity to mitigate the effects of impurities in the cooling water through the condenser or corrosion products. Thus, when condenser leakage occurs, the resultant impurities can enter the steam generators and cause severe changes in the pH, with resultant increases in corrosion rates. Although the three PWR vendors currently recommend AVT, both Westinghouse and CE had in the past recommended the use of phosphates to buffer impurities in their recirculating U-tube steam generators. * * *" Updated Experience Report, at p. 37.

Detection

"THE WITNESS: You see, the problem we have, sir, in this business of water chemistry, is even if your bulk water is perfect[,] on that localized basis where deposits come out, the whole thing could be blown right there. Because the bulk water could be perfect, immaculate, but on the localized basis your water chemistry could be totally different than it is in the bulk water. And therein lies the problem.

"EXAMINER WOLTER: By localized basis I assume you mean at the point where the corrosion is taking place.

"THE WITNESS: Or under a sludge deposit that sets on the bottom of the tube sheet.

"EXAMINER WOLTER: So really looking at this water chemisty, I take you answer at face value-- it doesn't really mean much.

"THE WITNESS: Not if you got crevices and deposits. This only works for the bulk water.

"EXAMINER WOLTER: And that's, of course, the important part, isn't it?

"THE WITNESS: The bulk --

"EXAMINER WOLTER: No, not the bulk, but where the corrosion takes place or the crevice is or where the deposit is, isn't that the critical area?

"THE WITNESS: That's the critical area.

"EXAMINER WOLTER: And you are telling us that there's no way to measure that water chemistry at that location?

"THE WITNESS: That's exactly -- in the crevices and under the deposits, that's correct.

"EXAMINER WOLTER: And there's no direct relationship or even indirect relationship, from your testimony, if I understand it correctly, between the bulk water and the water under thoe deposits or in those --

"THE WITNESS: I can haye anything in the water crevice area. The chemistry -- that can be totally different from this. And therein lies the problem.

"EXAMINER WOLTER: So what you are saying is that there's just no way we can tell whether the water chemisty at the point where it did the damage was similar or dissimilar to the chemistry that you have described in your Exhibit 68.

"THE WITNESS: In Exhibit 68 I addressed the bulk water chemistry

"EXAMINER WOLTER: That's right.

"THE WITNESS: And explained how the bulk water chemistry can go astray. But in Exhibit 68 I also explained how the localized water chemistry can be significantly different in the crevices and under deposits than it is in the bulk water.

"EXAMINER WOLTER: But there is no correlation necessarily between the bulk water chemistry that you have described in detail and the chemistry -- the water chemistry at the location where the corrosion takes place. There is no correlation between those two.

"THE WITNESS: No correlation between the two. Except we do know that in those deposit areas that the causticity of the environment has to be very, very high.

"EXAMINER WOLTER: In order for corrosion to take place.

"THE WITNESS: That's correct. See, this is part of the problem that we're into here today. As the steam generator tubes were tested certainly the Inco people and the Westinghouse people tested that material and tested it and experimented it in a wide variety of waters, but they certainly never expected to have a concentrated caustic solution exposed to the metal which in fact did occur under deposits and in crevices and in places of this nature.

"EXAMINER WOLTER: So am I correct, then, in assuming that the only way that you can tell whether the chemistry is bad at the point where the corrosion takes place is to wait until the corrosion actually takes place and then physically inspect periodically to see if corrosion is taking place?

"THE WITNESS: Basically, that absolutely correct. If I'm pulling a water chemistry off of the continuous blowdown, which WEPCO did, for example, very carefully and very continucusly, and they may be reading along and they say: Well, find, I have no free caustic present in the bulk water. But they may very well have free caustic present under the deposits.

"EXAMINER WOLTER: And there's no way they can tell that.

"THE WITNESS: There is no way to tell that. Because there's no way to get a water sample out of that area." Testimony Under Cross-Examination of Public Service Commission Witness, Dr. James R. Meyers, in Re Wisconsin Electric Power Company, Dockets 6630-ER-10 and 6630-UI-2, Transcript pp. 1775 to 1779

Feedwater Train

"A major disadvantage of AVT is that the boiler water is unbuffered and subject to extensive and rapid pH excursions in the event of feedwater contamination. Further, excessive hydrazine can decompose producing ammonia; unacceptable amounts of ammonia in the condenser or feedwater train can corrode copper-base alloys and allow potentially-deleterious corrosion products to enter the steam generator." Testimony of Public Service Commission Witness, Dr. James R. Meyers, in Re Wisconsin Electric Power Company, Dockets 6630-ER-10 and 6630-UI-2, at Exhibit 68, p. 17.

"Until better control of the feedwater chemistry is established, we suggest that samples from the discharge of the high pressure feedwater heaters be monitored frequently for iron and copper. A review of steam generator crud analysis from Units 1 and 2 sampled February, 1973 shows the presence of Cu in steam generator crud at levels substantially higher than six other operating plants reviewed.

"As discussed in the previous chemistry review continued operation with elevated oxygen and ammonia concentrations in the feedwater presents a potentially deleterious chemical environment to this copper and ferritic system. Accelerated corrosion can occur under these conditions to the feedwater reheat system. These corrosion products would be transported to the steam generator thus contributing to the formation of sludge on the tube sheet." Letter from G. W. Hood(W) to G. A. Reed(WE), dated February 4, 1974, re Point Beach Chemistry Review, at p. 2.

SIXTH CONTENTION
Operator Performance

Contention

An extremely high degree of operator performance is required both to properly maintain the proposed steam generators to prevent new corrosion and to respond to tube rupture accidents. Operator performance at Point Beach has seriously eroded in the past two years and no longer provides that necessary margin of

safety.

These problems with operator performance create a justiciable controversy as to whether the maintenance of the proposed steam generators will lead to continued tube degradation and as to whether operator response to tube rupture accidents will be adequate and thereby fail to comply with applicable Commission regulations, 10 C.F.R. §50.40(a) ("the health and safety of the public will not be endangered"), and 10 C.F.R. Part 50 App. A Crit. 14 ("pressure boundary shall * * * have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture"), such as to mandate denial of the operating license amendment.

<u>Basis</u>

> "There has been a discernable decline in the higher than average performance that had come to be expected of this utility * * * [and] there was a significant increase in the number of items of noncompliance. The increase is attributable to the lack of management attention and involvement necessary to maintain the discipline that has characterized the performance of Point Beach during previous years, and apparently stems from the loss of experienced personnel. This loss combined with the increased regulatory requirements and the more extensive maintenance caused by the steam generator tube corrosion have strained the licensee's resources. * * * During this evaluation period eight experienced people terminated inclding the Maintenance Supertendant, Superintendant of Chemistry and Health Physics, the Health Physicist, an Operations Supervisor, a Shift Supervisor and three experienced operators. In the prior evaluation period, eleven experienced management people or engineers had left; thus, the only management position not to be recently vacated is that of the Plant Manager. The personnel lost were replaced by promotion, depleting the overall experence level." Systematic Assessment of Licensee Performance, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, dated June 1982, at Transmittal Letter p. 2 and Enclosure pp. v. and 5.

SEVENTH CONTENTION
Unspecified Problems with Proposed Steam Generators

Contention

The proposed Model F Westinghouse steam generators may be expected to experience new forms of tube degradation of an undefined nature that cannot be specifically anticipated at this time, just as first the Model 51 and later the Model D steam generators, which succeeded the existing Model 44 steam generators, experienced new and unanticipated forms of degradation.

This inability to anticipate the entire scope of potential problems creates a justiciable controversy to support any inquiry reasonably related to whether the proposed steam generators will continue to suffer from tube degradation, and thereby fail to comply with applicable Commission regulations, 10 C.F.R. §50.40(a) ("the health and safety of the public will not be endangered"), and 10 C.F.R. Part 50 App. A Crit. 14 ("pressure boundary shall * * * have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture"), such as to mandate denial of the operating license amendment.

Basis

> "The small radius U bends in the inner (or first) row of tubing in [model 51] Westinghouse steam generators [which succeeded the model 44 generators at Point Beach] have been subjected to primary-side-initiated stress corrosion cracking. These cracks have occurred either at the apex of the U-bends or at the tangent point transition between the U-bend and the straight span portion of the tubing. At domestic plants such as Surry Units 1 and 2 and Turkey Point Unit 4, apex cracks have occured as a result of service-induced ovality of the tube as a result of the denting process. * * *
>
> "Apex cracks have also been observed in at least two

Westinghouse-designed foreign facilities. Doel, in Belgium, experienced a large leak at the apex of an inner row U-bend. Although there was no active denting at this unit that the staff is aware of, there was significant ovality of the tubing, which was believed to have been introduced during the fabrication process. * * *

"Another category of U-bend cracks includes stress corrosion cracks located in the transition area between the U-bend and the straight portion of the tubing. These cracks have generally been observed at plants which have not experienced denting. This tangent point cracking phenomenon has been responsible for numerous small leaks over the past three years affecting Westinghouse Model 51 steam genrators, particularly those at Trojan Unit 1.

" * * *

"It is believed that the 'opposite side' transition geometries [where the cracks have been located] were introduced during the fabrication process and resulted in increased residual stress at this location. The fabrication procedure includes the insertion of an internal ball mandrel through the U-bend during the bendin process to prevent excessive tube ovality. Westinghouse has reviewed the bending techniques used by [its] S[pecialty] M[etals] D[ivision] during the period in which U-bends exhibiting opposite side transitions were fabricated. Westinghouse has been unable to date, to identify exactly why certain tubes were affected and others were not." Updated Experience Report, at p. 13.

"Ringhals Unit 3, a three-loop Westinghouse plant in Sweden [with Model D steam generators which succeeded the Model 44 generators], was shut down on October 21, 1981 because of a 2.6 gpm primary-to-secondary leak. * * * The steam generators, W preheat type (Figure 2), are similar in design (Model D) to those at McGuire Unit 1, the only domestic operating plant with this type of steam generator.

"The leaking tube was located within the preheater section on the cold leg side of the steam generator. The ECT results revealed numerous tubes with ECT indications localized within the preheater section at baffle plate locations. * * *

" * * *

"Westinghouse believes the ECT indications are attributable to excitation of the steam generator tubes from high fluid velocities and that the tube walls are being worn down from vibrational rubbing against baffle plates in the preheater sections of these steam generators. Westinghouse further believes that a reduction of flow velocity by controlling total feedwater flow should reduce the potential for vibration." Updated Experience Report, at p. 16.

DATED at Madison, Wisconsin, this 5th day of November, 1982.

WISCONSIN'S ENVIRONMENTAL DECADE, INC.

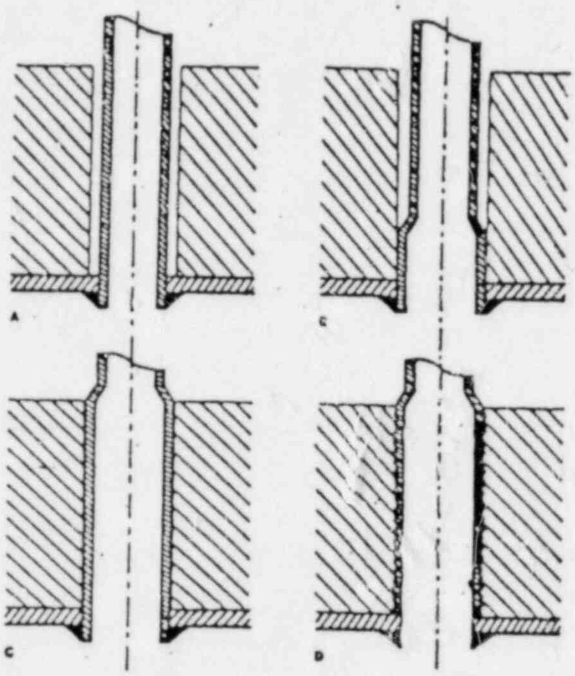
by

PETER ANDERSON
Co-Director

114 North Carroll Street
Suite 208
Madison, Wisconsin 53703
(608) 251-7020

ATTACHMENT 1

Diagram of Differing Tube Expansion Techniques



A welded
B partly rolled in and welded
C completely rolled in and welded
D explosion bonded and welded

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

Wisconsin Electric Power Company
POINT BEACH NUCLEAR PLANT UNIT 1
DOCKET NO. 50-266
Operating License Amendment 2
(Steam Generator Replacement Proceeding)

AFFIDAVIT OF PETER ANDERSON

STATE OF WISCONSIN)
) ss.
COUNTY OF DANE)

PETER ANDERSON, being first duly sworn, on oath states:

1. He is Co-Director of Wisconsin's Environmental Decade, Inc., a party in the above-captioned matter.

2. He prepared Decade's Contentions Concerns Steam Generator Replacement, dated November 5, 1982.

3. The excerpts cited in the Basis section of the Decade's Contentions Concerning Steam Generator Replacement are, to his own knowledge and belief, true and correct quotations from the documents indicated therein.

Subscribed and sworn to before me this 5th day of November, 1982.

Notary Public
State of Wisconsin
My commission expires 3/86

WED-PA-11/05/82-P:50266NRC.P53-5

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Wisconsin Electric Power Company
POINT BEACH NUCLEAR PLANT UNIT 1
Docket Nos. 50-266 OLA-2
CERTIFICATE OF SERVICE

I certify that true and correct copies of the foregoing document will be served this day by depositing copies of the same in the first class mails, postage pre-paid and correctly addressed, to the following:

Peter B. Bloch, Chairman
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Hugh C. Paxton
1229 - 41st Street
Los Alamos, New Mexico 87544

Dr. Jerry R. Kline
Atomic Safety & Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docketing & Service
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Richard Bachmann
Office of Executive Legal Director
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Bruce W. Churchill
Shaw, Pittman & Potts
1800 M. Street N.W.
Washington, D. C. 20036

Carol Pfefferkorn

Carol Pfefferkorn

Date: 11-5-82

NOTE: The mailing here is Federal Express