WED: 12/17/79

UNITED STATES OF AMERICA

BEFORE THE NUCLEAR REGULATORY COMMISSION

POINT BEACH NUCLEAR PLANT UNIT 1
DOCKET NO. 50-266

MEMORANDUM IN SUPPORT OF
REQUEST BY WISCONSIN'S ENVIRONMENTAL DECADE, INC.
FOR HEARING ON CONFIRMATORY ORDER

I. INTRODUCTION

Point Beach Nuclear Plant Unit 1 which commenced commercial operation in December 1970, has been observed to experience steam generator tube degradation due to caustic stress corrosion, wastage and denting since April 1971. (See: Wisconsin Electric, Current Report to the Securities & Exchange Commission, Form 8-K, dated October 25, 1979, at 2.)

Inservice inspections of the facility in August and October 1979 disclosed extensive acceleration of tube degradation. (See: Office of Nuclear Reactor Regulation, Confirmatory Order for Modification of License, dated November 30, 1979, In the Matter of Wisconsin Electric Power Company, Docket 50-266, at 1.)

On November 14, 1979, Wisconsin's Environmental Decade, Inc., petitioned the Commission for an order prohibiting reopening of Point Beach 1 and for an adjudicatory hearing on the sufficiency of the agency's bases of continued operation of nuclear reactors experiencing significant tube degradation. (See: Petition of Wisconsin's Environmental Decade, Inc., dated November 14, 1979, In the Matter of Wisconsin Electric Power Company, Docket 50-266.)

On November 28, 1979, the Commission met to consider the Decade Petition and hear its Staff recommendation.

On November 30, 1979, the Office of Nuclear Reactor Regulation denied the Decade Petition and authorized reopening of Point Beach 1 subject to certain conditions. (See: Confirmatory Order, op. cit.; and Director's Decision Under 10 CFR 2.206, dated November 30, 1979,

In the Matter of Wisconsin Electric Power Company, Docket 50-266.)

This Memorandum is submitted in support of the Decade's request for a hearing on the Confirmatory Order on the grounds that all necessary factors were not considered in the order and that critical new facts have subsequently occurred.

II. ARGUMENT

A. Point Beach Should be Closed Pending Resolution of the Tube Degradation Issue Being Experienced at the Plant.

The primary safety concern at issue involving the matter of steam generator tube degradation at Point Beach is the question of whether degraded tubes will rupture during the course of a loss of coolant accident causing steam binding and essentially uncoolable conditions in the reactor core. (See: Lewis, et al., "Report to the American Physical Society of the Study Group on Light Water Reactor Safety," 47 Review of Modern Physics 1, Summer 1975, at App. 1, S85.)

After a long period of discussion without resolution, in January 1979, the Staff set forth a program for further analysis of the tube degradation issue and recommended continued operation of plants experiencing this problem on six ennumerated bases pending final resolution. (See: Eisenhut, et al., Summary of Operating Experience with Recirculating Steam Generators, Nuclear Regulatory Commission (NUREG-0523) January 1979, at 46.)

In our previous filing, we refuted each of the six generalized bases for continued operation, (see: Petition, op. cit., at 6-8) and,

to date, no formal response has been put forward by anyone disputing this refutation insofar as it relates to tube degradation in the free standing region of the tube bundle.

Instead the basis for continued operation of Point Beach is promised in the Staff's evaluation on three primary grounds:

First, the licensee's proposed "package" of new procedures in addition to those set forth in the technical specifications (including but limited to repeated crevice flushing, increased eddy current testing, periodic hydrostatic testing, improved tolerances for primary-to-secondary leakage, modification of primary pressure and temperature and closer monitoring of condensor in-leakage) will provide adequate assurance of identifying significant tube defects that may exist in operation;

Second, any further tube degradation will be confined to the region of the tube sheet where ruptures during a LOCA will be less likely to occur and leak rates will be substantially constrained by the sheet wall; and

Third, the realistic assumption of tube ruptures and secondary-to-primary leakage in the region of the tube sheet that might occur during a LOCA will be too low to give rise to concerns over steam binding and core melt. (See: Office of Nuclear Reactor Regulation, Safety Evaluation Report Related to Point Beach Unit 1 Steam Generator Tube Degradation Due to Deep Crevice Corrosion, November 30, 1979, at 17-25.)

For the reasons set forth below, we believe that there is inadequate

support to justify these three new bases for continued operation of Point Beach from necessary factors that were not considered and from new information and that an adjudicatory hearing is necessary to compile an adequate record on which to make a decision.

Therefore the portion of the Confirmatory Order permitting the licensee to resume operation of Point Beach should be reversed.

1. The Licensee's Proposed "Package" Does Not Provide Satisfactory Assurance that Significant Tube Defects Will be Identified Prior to Resumed Operation.

The Safety Evaluation Report states in regard to the licensee's proposed "package" as follows:

> The staff agrees that hydrostatic pressure tests prior to returning power and periodically during operation will provide a positive indication and increased confidence in steam generator tube integrity. . . . Similarly, the proposed decrease in the primary-to-secondary leak rate limit will provide conservative limits which will require timely plant shutdown and corrective actions. (See: Id., at 21.)

This is essentially a more subdued endorsement of the licensee's "package" that was given orally to the full Commission at its November 28, 1979 meeting. For two reasons we demur from this conclusion.

First, at that meeting, Commissioner Ahearne asked us whether the increased periodic hydrostatic pressure tests and more frequent eddy current tests assuaged our concerns. We responded and maintain here that these measure would only provide a significant degree of confidence if the rate of corrosion was low, which is not the case at Point Beach where the rate of degradation is the worst in the country. This is because the safety concern arises when there are a sufficient number of

incipient tube failures (whether one to ten per the American Physical Society or 100 to 200 per the Staff) at the moment of a LOCA to cause steam binding and a core melt. If the predicted rate of corrosion would leave a sufficient number of tubes degraded which might fail during a LOCA in the period between inspections, then those increased inspections will not provide an adequate margin of safety.

Also at the same meeting Commissioner Bradford asked us whether the reductions in the primary-to-secondary leak rates assuaged our concerns. We responded that this would only be effective _if_ the problem were _general_ corrosion which wears through the tube wall slowly and in which a small leak precedes failure, and not _stress_ corrosion which is characterized by weakening of the tube wall without a through wall crack until the system is subjected to sudden stress precipitating failures of the tubes. In Point Beach, it is stress corrosion which is the dominant problem at the present time (_see_: Safety Evaluation Report, _op_. _cit_., at 4-5), and therefore, lower primary-to-secondary leak rates will not provide any significant degree of assurance of anticipating all incipient failures during a LOCA. The vague reference by the licensee to some undefined laboratory tests purporting to show that small leaks will permit shutdown before critical cracking (_see_: Wisconsin Electric, Letter to NRC, dated November 23, 1979, at Enclosure 3, Viewgraph 21), is completely disproved by the plant's actual, real-life operating experience. Witness the February 1975 incident in which a 125 gpm primary to secondary leak occurred without warning and which

was the worst leak ever before experienced. (See: NRC, Report to the Congress on Abnormal Occurrences, June 1975, at A-1.) This is exactly what one would expect to occur under stress corrosion conditions.

After the close of that meeting, Mr. Case from the Staff came up to us and stated that they concurred in our response as to the adequacy of the licensee's package. (See: Affidavit of Peter Anderson, dated December 17, 1979 which is attached hereto and incorporated herein by reference.) However, they did not see fit to inform the Commissioners of this fact or to modify the Safety Evaluation Report to clearly indicate that whatever "increased confidence" is provided by periodic hydrostatic testing and decreased primary-to-secondary leak rate limit will be sharply mitigated by virtue of the very high rate of tube degradation and the predominance of stress corrosion.

Be that as it may, we do not believe that one can responsibly dispute that the licensee's proposed "package" is of marginal value because of the particular facts of the Point Beach experience. Essentially, we believe that the real basis for continued operation on the part of the Staff is not any significant assurance from the so-called "package" but rather from the alleged confinement of tube degradation to the tube sheet. (See: Id.)

Second, notwithstanding the representations of the Staff and the licensee that the proposed "package" would minimize or substantially eliminate further leaks between inspections, a 200 gpd leak in the "B" steam generator of Unit 1 at Point Beach suddenly erupted the evening

of December 11, 1979, barely 11 days after beginning to return to service and less than one week at 77% to 80% of full power. (Note: The Licensee Event Report for this occurrence has not yet been filed.) It would appear at this point in time that corrosion is not only not abating, but rather is accelerating at an even faster rate, notwithstanding the "package." Alternatively, the quality controls in the inspection process were inadequate at the same time when the licensee made an all-out effort, that will not oft be repeated, to insure that there were no mistakes. In either case subsequent events conclusively demonstrate that the "package" is ineffective.

2. Tube Degradation Has Not Been Confined to the Tubesheet.

The Safety Evaluation Report states in regard to the alleged confinement of present and projected tube degradation to the region below the tubesheet:

> Subsequent operating experience at Point Beach Unit 1 (since February 1975) indicates that the wastage and caustic SCC phenomenon above the tubesheet have essentially been arrested. (See: Id., at 4.)
>
> The most recent concerns regarding the integrity of steam generator tubes at Point Beach Unit 1 involve corrosion damage to tubes within the thickness of the tubesheet. (See: Id., at 5.)
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> No crevice indications extending above the tubesheet have been observed to date. (See: Id., at 10.)
>
> Because the deep crevice cracking is peculiar to the local chemistry conditions in the tube to tubesheet crevice, the phenomenon will be limited to that area. This is confirmed by the location of all defects which have been observed during inservice ECT inspection, and by the laboratory examinations and mechanical testing of tube samples removed from Point Beach Unit 1. (See: Id., at 20.)

This, combined with the third point discussed below concerning lower leak rates within the tubesheet, appear to be the real Staff justification for continued operation. For two reasons, the attempt to find safe operation from assumed tube degradation confined to the tubesheet are unfounded.

First, there is an overriding matter in this regard of the utmost gravity that casts a shadow on the integrity on the representations of the Staff itself.

We have just learned of the existence of a letter and attached Licensee Event Report, dated November 16, 1979, from Wisconsin Electric to the Commission, and received by the Commission apparently on November 19, 1979, before the Commission's November 28, 1979 meeting. The report indicates that four of the tubes which the eddy current test showed to be defective were defective at the "top of tubesheet" and one was "one-half inch above tubesheet." (See: Wisconsin Electric, Licensee Event Report No. 79-017/01T-0, dated November 16, 1979, at Tables 1 and 2.)

Notwithstanding the fact that the Staff had this document from at least November 19, its representations to the Commission on November 28 maintained that all degradation was in the crevice where leakage would be throttled and allegedly no safety problem would exist. Indeed, when Commissioner Ahearne asked the Staff what percentage of the defective tubes had the defect within 0.15 inch of the top of the tubesheet (where the tube could pull out of its sleeve if circumferentially cracked), the Staff responded that it did not have that data and the

licensee did not proffer the information.

Whatever the Staff's and licensee's motivation, the effect of their withholding the damaging information in their possession was to mislead the Commission in a matter of the most fundamental importance to the decision.

It is clear that there are no further grounds for maintaining the ruse that degradation will be confined exclusively to the tubesheet.

Second, this latest information confirms our previous representations on the question of whether corrosion will remain in the crevice. For it is most important to understand that the assertions claiming that the only remaining problem is deep crevice corrosion is a hypothesis and not a fact. We have shown above that it is not even true as a matter of fact that past corrosion is within the tubesheet. Beyond that, to have any confidence in a hypothesis that projects today's incorrect fact into the future, a rational person would require either a past track record of success in simple statistical extrapolations or a complete understanding of the problem. In point of fact, we have neither.

Tube degradation has continued to change in location, form and extent without warning. The licensee conceded in hearings before the Public Service Commission of Wisconsin that it did not anticipate the movement of the previously experienced wastage and caustic stress corrosion above the tubesheet to deep crevice corrosion or the rapid acceleration of degradation. (Note: The transcript has not yet been

prepared.) And the Safety Evaluation Report itself concedes that as recently as January 1979, the Staff "did not consider deep crevice corrosion as it was not identified as a significant mode of tube degradation prior" to that time. (See: Safety Evaluation Report, op. cit., at 18.) The Report goes on to state that "the Point Beach Unit 1 situation is unique in terms of the extent and the rapid progression in the last twelve (12) months." (See: Id., at 5.) There is no rational basis, premised upon predicting the future from the past, for hypothesizing no tube degradation at the top of the tubesheet in the future.

Similarly, it follows that a complete understanding of the phenomena does not exist, since it was not predicted. We know that the encrusted caustic impurities extend from nearly two feet below the top of the tubesheet to a half of a foot or more above it. Although it may well be that the corrosive effect of these impurities might be intensified in the confined region of the crevice, that is no reason to conclude that no corrosion will occur at the top of the sheet. In telephone conversations with Staff, the only defense to this point is that we have not recently observed corrosion above the tubesheet--which brings the wheel around to where we started--without any responsible basis for pretending that the safety analysis can assume that no leakage will occur that is not in the crevice.

3. Steam Binding is an Undisputed Possibility for the Corrosion at Point Beach at the Top of the Tubesheet.

No one, to our knowledge, has disputed the validity of the American Physical Society's conclusion that tube ruptures in one to ten tubes

during a LOCA can cause steam binding and core melt--at least to the extent that the failures are above the tubesheet.

As indicated in the preceding section, substantial defects from caustic stress corrosion are now being found at the top of the tubesheet and above it. The Staff has adopted the licensee's conclusion that 40% of the thickness of the tube wall is "required to resist pressure loading during a LOCA" outside the tubesheet. (See: Safety Evaluation Report, op. cit., at 14.) Of the five tubes identified by in-plant eddy current testing as having defects at the top of the tubesheet during the October 1979 outage (approximately one month following the previous complete check of all inlet tubes), two had less than half of the required 40% of the wall thickness remaining to withstand LOCA pressure. (See: November 16, 1979 Licensee Event Report, op. cit., at Tables 1 and 2.) It is unknown how many additional tubes with defects of this magnitude at the top of the tubesheet were not identified by the in-plant eddy current test, especially since it is conceded that the test is not effective for intergranular corrosion without through-wall cracking which is pervasive at Point Beach in the tubesheet, (see: Safety Evaluation Report, op. cit., at 18 & 23), and which may extend to the boundary area at the top of the tubesheet.

Thus, there is a real possibility of an APS core melt situation in the event of a LOCA. The reasonable assurance of safety in the operating license for Point Beach is premised upon the unit's being able to withstand a double-ended pipe break. That assurance can no

longer be responsibly found to exist.

4. Even if Further Tube Degradation Were Confined to the Tube Sheet, There Still Exists a Serious Safety Problem.

It should be unmistakably clear that the facts make untenable any claim that corrosion is confined exclusively to the crevice. But, even if, arguendo, it were, there still exists a serious unresolved safety issue that vitiates the basis for continued operation.

The Safety Evaluation Report states, based upon an incorrect assumption that all tube ruptures during a LOCA would be within the tubesheet that:

> A large number of tube failures (collapses) would therefore be necessary before the secondary to primary leak rate would result in steam binding and adversely affect the ability of the ECCS to cool the core. (See: Safety Evaluation Report, op. cit., at 21.)

Underlying calculations in the Report imply that approximately 185 tubes with a cumulative secondary-to-primary leak rate of more than 1300 gpd would have to rupture during a LOCA to cause steam binding per the conclusions of the APS--if all of the failures are within the tubesheet.[1]

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[1] The Staff's calculations have never been subjected to adjudication, and it should be clear that we dispute them. One of the problems with their calculations is that it appears they have not included synergistic effects that will retard reflood rates to unacceptable levels. In particular, we are concerned about the issue raised in the same APS report about flow blockage from ruptured rods during a LOCA which make reflood rates less than one inch a second too low to be allowed. (See: Report of the American Physical Society, op. cit., at App. 1, S91.) This should be compared to the APS's previous statement that reductions in reflood rates from the other factor of steam binding can be to as low as one inch per second (see: Id., at App. 1, S90), thus making cooling of the core further questionable.

(See: Id., at Appendix A.) This difference with the APS one to ten tubes is accounted for by attributing a constraining effect on in-leakage from the tube wall.

Apparently, the Staff discounts the possibility of problems occurring with 100 to 200 tubes resulting in 1300 gpd in leakage. Had the Staff carefully evaluated the actual circumstances at Point Beach, instead of automatically dismissing the matter, it would have found that conditions have deteriorated to such an extent as to make these numbers all too credible.

Three readily identifiable sources of this leakage must be considered. One concerns tubes with defects not next to the top of the tubesheet for which the constraining effect may apply (i.e. more than 0.15 inch from the top), the second concerns tubes which are next to the top for which the constraining effect will likely not apply (i.e. within 0.15 inch from the top), and the third, leaking plugs used to seal previously identified defective tubes.

First, as to those tubes with defects not next to the top, the salient question is the number of those tubes with sufficient corrosion to fail during a LOCA. The Staff has apparently adopted the licensee's conclusion that 10% remaining wall thickness is required to prevent a double ended tube failure during a steam line break (see: Safety Evaluation Report, op. cit., at 13-14), but not the licensee's hypothesis that "collapse within tubesheet [is] not possible due to confinement" during a LOCA. (See: Wisconsin Electric, Letter to the NRC, dated

November 23, 1979, at Enclosure 3, Viewgraph 19, emphasis added.) Instead, the Staff concludes, without any stated basis, that "[t]ube collapse during LOCA is highly unlikely since tube ovalization during collapse would be constrained by the tubesheet." (See: Safety Evaluation Report, op. cit., at 14, emphasis added.)

Again, rational discussion is inhibited by the failure of the Commission to compile a complete record on this fundamental issue that would provide a reliable minimum wall thickness, with an adequate margin of safety, for analyzing which tubes with a defect in the tubesheet will not survive a LOCA. In addition to the failure to quantify what "highly unlikely" means, there is a substantial problem with assuming a given event can be defined as having only one cause--in this case by assuming collapse only comes from ovalization. The possibility of tube collapse during LOCA from a longitudinal defect dovetailing open inward, as just one example, is conveniently ignored, and thereby pretended not to exist.[1]

Some proxy for this parameter of minimum wall thickness must be used to prevent the absence of all consideration from wishing away a very real problem. For the sake of discussion, we use a value of 20%

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[1]This unshakable preoccupation in the Staff and the entire nuclear industry with blithely assuming given events can be entirely explained by a single cause has been identified as one of the reasons why the Commission did not anticipate a Three Mile Island type of accident. Coremelts were just assumed to only be caused by catastrophic sudden failures like a double ended pipe break and not by less important small breaks. (See: President's Commission on the accident at Three Mile Island, The Need for Change (1979) at 9.)

of remaining wall thickness, and compare that value to the combined experience of the August and October outages because they occured so close in time.

The reports from those outages disclose that 177 tubes were identified by the eddy current test as having defects 80% of more of the wall thickness out of a total of 230 tubes with any identified defect. (See: Wisconsin Electric, Licensee Event Report, Nos. 79-012/01T-0, 79-013/01T-0 and 79-017/01T-0, dated August 20, 1979, August 31, 1979 and November 16, 1979.) Based upon the evidence available, it must be concluded that these numbers represent only a portion of the actual total number of tubes so afflicted.

Westinghouse Electric Corporation conducted detailed laboratory testing on three tubes from Point Beach after the October 1979 refueling, one tube which did have an eddy current test indication of a defect and two tubes which had no such indication. Both of the ostensibly good tubes were shown with laboratory metallurgical analysis to be significantly defective, 50% and 33%, respectively. (See: Safety Evaluation Report, op. cit., at 10-12.)

At the November 20, 1979 meeting of the Staff, licensee and petitioner, Westinghouse attempted to explain the failure to identify defects in the two good tubes with in-plant eddy current tests by hypothesizing that the defects found in the laboratory were not caused by corrosion in the steam generator, but instead by the pulling force exerted to remove the tube for laboratory testing, a hypothesis which has been adopted by the Staff. (See: Safety Evaluation Report, op. cit.,

at 12). During that November meeting, Westinghouse stated in support of this hypothesis that the eddy current tests run in the laboratory showed the same defect as disclosed by the metallurgical analysis, implying that the defect was not identified in the in-plant ECT because it did not occur until removed.

Of course, even if, arguendo, this were true (which it is not), a laboratory run ECT conducted by a highly trained professional without background noise from the tubesheet is not comparable to an in-plant ECT conducted in the radioactive plant by a less well-trained employee and with tubesheet interference. Be that as it may we insisted on seeing the laboratory ECT results to corroberate this assertion, and found that the entire claim was apparently fabricated from whole clothe.

Although the licensee promised to provide to us the laboratory results at the November meeting, it was not forthcoming and several follow-up telephone calls on our part were required. It turned out that the delay was caused by the fact that laboratory eddy current tests had not in fact been run at the time of the November meeting. More disturbing, the post hoc results of the lab tests subsequently performed did not bear out the hypothesis. One of the tubes which had no in-plant ECT and also had no laboratory ECT, had a 33% defect with the mettalurgical analysis. (See: Personal Communication with David G. Porter on December 13, 1979, concerning tube R20C73 in "A" Generator.)[1]

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[1]The portentious implications of a laboratory ECT failing to detect a 33% defect in a tube when there was no tubesheet to cause interference

This is one more graphic illustration of the necessity of an adjudicatory hearing to test the welter of assertions. Even if, arguendo, one were to postulate some other hypothesis to resurrect the validity of the eddy current test, there would be no lessening in the safety concerns. This is because of the fact that if one assumes an effective ECT, then one still has to explain the rapid appearance of cracked and defective tubes--in the case of the most recent February 11, 1979, outage, a tube going from 0% defect to 100% defect in the space of 11 days, five of which were slowly ramping up and the remaining 6 days were at only 77%-80% of full power. Indeed, the safety implications of an assumed effective ECT, but runaway corrosion, is probably even more ominous.

In any event, the available facts demonstrate a realistic probability that 185 tubes could fail during a LOCA and cause steam binding even without any further sources of in-leakage.

Second, as to tubes near the top of the tubesheet, the Staff acknowledges that the constraining effect of the tubesheet throttling leakage that might cause steam binding may not operate for cracks within 0.15 inch of the top of the tubesheet because the entire tube might pull out of its fitting if the crack is circumferential. This could create an unrestrained gusher. (See: Safety Evaluation Report, op. cit., at 14.)

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should not be underestimated in terms of the in-plant test's validity even in free standing regions of the generator.

Inexplicably, the Staff does not pursue the extent of degradation in this upper region of the tubesheet on the basis of the unsupported statement "leak rates will be large enough to allow detection during normal operation." (See: Id.)

Previously, it has been shown that this kind of unsupported statement is borne out neither by the actual operating history of the plant nor by what one would expect when the problem is caustic stress corrosion and not general corrosion. Thus, this potential additional source of in-leakage should be evaluated.

Looking, then, at the same data from the August and October 1979 outages, 40 tubes had defects identified as above one inch of the top of the tubesheet (see: Licensee Event Report Nos. 79-012/1T-0, 79-013/01T-0 and 79-017/01T-0, dated August 20, 1979, August 31, 1979 and November 15, 1979), and undoubtedly many more such defects were not picked up. We have asked the Staff to direct the licensee to provide the exact elevation data for these tubes to determine which were within 0.15 inch, but were refused. If one assumes a normal distribution of these 40 tubes with a mid-point mean and standard deviation, that would imply that slightly more than three are within 0.15 inch of the top. While the 40% remaining minimum tube wall thickness to withstand a LOCA in the free standing region is not necessarily applicable to this boundary area, neither is the 20% proxy value for deep in the crevice. A mid-range 30% value may be used for the purpose of discussion. Again, using an assumption of a normal distribution, there would be slightly

less than three tubes within 0.15 inch of the top of the tubesheet with 30% or less remaining thickness in the tube wall. Even if the tube does not completely pull out of the tubesheet, assuming the crack is longitudinal and not circumferential, one cannot reliably state that the full throttling effect will occur either.

At this point, any further attempt to quantify would not be productive until an adjudicatory hearing could be held to parse out the exact number. But the discussion does demonstrate another potential source of significant in-leakage during a LOCA and the necessity for resolution of this question.

Third, no one has considered the question of leaking plugs. There can be no dispute that plugging a tube does not completely end leaking. During the October 1979 outage, five plugs previously inserted were found to be leaking and had to be welded closed. (See: Wisconsin Electric, Licensee Event Report, No. 79-017/01T-0, dated November 16, 1979.) Yet there has been no analysis put forward to estimate potential in-leakage into the primary side during LOCA from faulty plugs.

Clearly, even if, arguendo, it were incorrectly assumed that no corrosion would occur outside the tubesheet, there is a woefully insufficient basis for continued operation.

B. An Adjudicatory Hearing is Essential on the Generic Steam Generator Tube Degradation Problem.

The Atomic Energy Commission inaugurated the era of commercial nuclear reactors without first, among other things, conducting a full-scale review of acceptance criteria for the emergency core cooling

system. Not until June 1971 were interim acceptance criteria promulgated (see: 36 Fed. Reg. 12, 247), and then without any hearing or public comment, apparently in an attempt to forestall the issue being raised in pending licensing proceeding.

After a significant amount of public pressure, in November 1971 the Commission's predecessor agency commenced a rule-making proceeding on the interim criteria in which the ability to conduct a full and searching review were severely circumscribed by barring the right of discovery. (See: In the Matter of Interim Acceptance Criteria for Emergency Core-Cooling Systems for Light Water Cooled Nuclear Power Plants, Docket RM 50-1.)

In that proceeding, the Union of Concerned Scientists attempted to argue that:

> (T)he rupture or failure of only a very few, a handful of the steam generator tubes in a PWR under LOCA conditions can inject sufficient steam to stall totally the reflood capability (after a LOCA) and so to insure clad melting. It appears likely, if not certain, in view of the known aggravated corrosion of and wall thinning in steam generator tubes in several operating reactors, that the forces developed by a cold-leg pipe break would rupture sufficient tubes to cause this stalling. (See: Union of Concerned Scientists, An Assessment of the Emergency Core Cooling Systems Ruling-making Hearing (1973), at 5.49.)

But the Hearing Board precluded the issue from consideration.

Two years later, in the American Physical Society Review of the nuclear safety issue, the respected review panel, corroberated the USC findings and concluded in this regard:

> (I)t appears that rupture of a few tubes (on the order of one to ten) dumping secondary steam into the depressurized

> primary side of the reactor system could exacerbate steam binding problems and induce essentially uncoolable conditions in the course of a LOCA for PWR's with ECCS of current design. . . . Thus the potential for steam generator tube leakage appears to be a serious problem which was precluded from evaluation at the ECCS hearings. (See: Report to the American Physical Society, op. cit., at App. 1, S-85-91.)

Still, no review of the issue was forthcoming as the Reactor Safety Study also omitted reference to the serious problem. (See: Rasmussen, Reactor Safety Study, WASH-1400 (1975). And, again, the agency was criticized, this time by the review group which it had established, for continuing to ignore the issue. (See: Ad Hoc Review Group, Risk Assessment Review Group Report to the Nuclear Regulatory Commission, NUREG/CR-0400 (1978), at 48.)

Yet, at this writing, the only thing that the Staff has done is issue a report outlining an area of future study and providing six bases for continued operation while the study is conducted. (See: Eisenhut, op. cit., at 46.) With the Decade Petition having established that those bases were unfounded, a new basis was adopted that the foregoing discussion demonstrates was premised upon ignoring all factors which detract from the predetermined conclusion and upon the withholding of fatal information from the full Commission and this intervenor.

It is time, at long last, for that full inquiry of the tube degradation issue to be commenced with an adjudicatory proceeding.

It is time, at long last, for the Commission to reverse its reputation for placing "the industry's convenience" above its "primary role of assuring safety." (See: The Need for Change, op. cit., at 19.)

III. CONCLUSION

For the foregoing reasons, the portion of the Confirmatory Order permitting restarting Point Beach 1 should be reversed and an adjudicatory hearing should be commenced on the generic tube degradation issue.

Respectfully submitted by,

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Dated: December 17, 1979