859

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
Sheldon J. Wolfe, Chairman
Dr. David L. Hetrick
Dr. James C. Lamb, III

In the Matter of	)	
	)	NRC Docket No. 50-289 OLA
METROPOLITAN EDISON COMPANY, <u>ET AL</u>.	)	(ASLBP Docket No. 83-491-04 OLA)
	)	(Steam Generator Repair)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	October 31, 1984

INITIAL DECISION
(Amendment To Operating License)

Appearances

George F. Trowbridge, Esq., Bruce W. Churchill, Esq.,
Diane E. Burkley, Esq., and Wilbert Washington, II, Esq.,
for Metropolitan Edison Company

Joanne Doroshow, Esq. and Louise Bradford,
for Three Mile Island, Alert, Inc.

Thomas Y. Au, Esq.,
for the Commonwealth of Pennsylvania

Mary E. Wagner, Esq.,
for the U. S. Nuclear Regulatory Commission

DS02

INITIAL DECISION

TABLE OF CONTENTS

Page

OPINION

I. INTRODUCTION[1]

A. BACKGROUND

1. Steam Generators' Description.

Three Mile Island Nuclear Station, Unit No. 1 (TMI-1), located in Dauphin County, Pennsylvania, is a 776 megawatt pressurized water reactor having two vertical, straight tube and shell, once-through steam generators ("OTSG"). Each steam generator contains 15,531 Inconel-600 tubes. Each tube is 56 feet, 2-3/8 inches in length, with a 0.625-inch outer diameter and a 0.034-inch minimum wall thickness. The ends were inserted into holes drilled in two 24-inch thick carbon steel tubesheets at the top and bottom of the steam generator. The tube was fully inserted, and protrudes about 1/2-inch beyond the upper face of the Inconel clad upper tubesheet and the lower face of the lower tubesheet, into the primary head at each end of the steam generator. There is a nominal 0.005-inch radial gap between the outer surface of the tube and the surface of the tubesheet hole. During manufacture of the steam generators, the tubes were sealed to the tubesheet at each end by rolling to a depth of about 1-1/4 inches, and welding on the primary side of the tubesheet surface. Primary coolant (at a pressure of about 2200 psig) flows within the tubes, and secondary system water and steam

---

1 Part I sets forth certain uncontested facts.

(at a pressure of about 950 psig) are heated outside the tubes. Thus the tubes, including the seal at each end, constitute part of the reactor coolant pressure boundary between the primary and secondary systems.

TMI-1 has been shut down since its last refueling outage in 1979 pending the outcome of restart proceedings before the Nuclear Regulatory Commission relating to the accident at TMI, Unit 2 which occurred on March 28, 1979. In November 1981, primary-to-secondary leakage was discovered during testing of the TMI-1 reactor coolant system. This leakage was caused by intergranular stress assisted cracking of steam generator tubes. Eddy current testing (ECT) revealed that 95 percent of the defects occurred within the top seven inches of the upper tubesheet (UTS).

2. Description of the Kinetic Expansion Repair Process.

Of the 31,062 tubes in both steam generators, 29,838 were repaired by kinetically expanding the tubes within the tubesheet to provide a new seal to the tubesheet below where the defects were detected. This was done by detonating an explosive cord encased in a polyethelene insert which had been placed into the tube. The resulting explosive energy was transmitted to the tube wall by the polyethelene insert, pressing the tube against the tubesheet. The tubes were expanded from the top of the upper tubesheet down either 17 inches or 22 inches, depending on the elevation of the lowest ECT indication within the upper tubesheet. This provided a 6-inch or greater ECT indication-free expanded length between the lowest elevation ECT indication and the bottom of the expansion to serve as the new pressure boundary.

3. Proceedings.

On May 9, 1983, the Licensee submitted to the Nuclear Regulatory Commission an application for an amendment to its operating license requesting that it be permitted to revise the technical specifications to recognize steam generator tube repair techniques, other than plugging, and that the Commission approve the proposed kinetic expansion repair technique used at the facility. On May 31, 1983, at 48 Federal Register 24231, the Commission published a notice captioned "Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing."[2]

In a Memorandum and Order of November 29, 1983, LBP-83-76, 18 NRC 1266, as amended by the Order of December 1, 1983 (unpublished), the Board admitted as intervening parties Three Mile Island Alert, Inc.

---

[2] In a letter dated January 13, 1984, the Staff advised the Board and the parties that, at a meeting of the Commission on January 10, 1984, the Commission considered the question whether to concur in the Staff's proposed final no significant hazards consideration determination for the TMI-1 steam generator repair license amendment. The Staff also stated that, after voting 2-2 on the question, with one Commissioner not voting, the Commission then stated that no action should be taken by the Staff to issue the final determination or the amendment until the Commission had voted again and reached a decision on the matter.

(TMIA) and the Joint Intervenors (Ms. Jane Lee, Mr. Norman Aamodt)[3] and admitted certain subparts of their contentions.

Thereafter, in a Memorandum and Order of June 1, 1984 (unpublished), the Board granted the Licensee's and the Staff's motions for summary disposition of the Joint Intervenors' contentions, and dismissed Joint Intervenors as a party. The Board granted in part and denied in part the Licensee's and the Staff's motions for summary disposition of TMIA's contentions. With respect to TMIA's Contentions 1.a and 1.b, the contentions which were not entirely dismissed, the Board identified specific issues as to which evidence was to be presented at the hearing. These issues are discussed below in Part II.

The evidentiary hearing was held on July 16-18, 1984.[4] The Licensee, the Staff and TMIA participated, as well as the Commonwealth of Pennsylvania which, on July 9, 1984, had filed a motion for leave to participate as an interested State pursuant to the 10 C.F.R. § 2.715(c). Only the Licensee and the Staff presented witnesses.

The Licensee, the Staff and TMIA filed proposed findings of fact and conclusions of law -- the Commonwealth of Pennsylvania did not.

---

3 A third joint petitioner, Dr. Bruce Molholt, withdrew his petition for leave to intervene during the course of the special prehearing conference held on October 17, 1983.

4 Limited appearance statements were also taken during the course of the hearing.

B. CONTENT OF OPINION AND FINDINGS

Part II of this Opinion discusses and resolves the contentions. Part III reflects our conclusions. The Board's underlying Findings of Fact and Conclusions of Law are appended and are incorporated by reference. An Order is also appended.

It should be noted that all of the proposed findings of fact and conclusions of law submitted by the parties that are not incorporated directly or inferentially in this Initial Decision are rejected as unsupported in law or fact or as unnecessary to the rendering of this Initial Decision.

II. CONTENTIONS

A. CONTENTION 1.a[5] (Fdgs. 1-43

Issue 1.a Reliability of Leak Rate Measurements (Fdgs. 2-20).

---

5 TMIA's Contention 1.a, as originally admitted, alleged with respect to the kinetic expansion repair technique that "post repair and plant performance testing and analysis . . . and proposed license conditions are inadequate to provide sufficient assurance that tube ruptures . . . will be detected in time and prevented . . . ." As noted in Part IA, above, in the unpublished Memorandum and Order of June 1, 1984, at page 23, the Board denied in part the Licensee's and the NRC Staff's motions for summary disposition of Contention 1.a, and identified seven issues with respect to which evidence was to be presented at the hearing.

The Staff's proposed License Condition 4, as modified in Supplement 1 to the Safety Evaluation Report (SER), reads as follows:

> The Licensee shall confirm the baseline primary-to-secondary leakage rate established during the steam generator hot test program. If leakage exceeds the baseline leakage rate by more than 0.1 GPM, the plant shall be shut down and leak tested. If any increased leakage above baseline is due to defects in the tube free span, the leaking tube(s) shall be removed from service. The baseline leakage shall be re-established, provided that the present Technical Specification limit of 1.0 GPM is not exceeded (SE Section 3.3).

The Board requested evidence on the above-captioned issue because the proposed license condition on leak rate limitations might not be effective if the measurements of leak rates were not sufficiently reliable. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

The Licensee determined the baseline primary-to-secondary leakage to be 0.02 GPM during the steam generator hot test program. The facility is to be shut down if leakage increases by 0.1 GPM above the baseline, i.e., if the leak rate reaches 0.12 GPM. This may be compared with the existing leak rate limit of 1.0 GPM. Subsequent tests may increase the baseline, provided that the limit of 1.0 GPM is not exceeded.

Statistical variations and measurement sensitivities are such that these limits are feasible. However, the most sensitive on-line instrument channel (the RM-A5L monitor of radioactive gas in the secondary system) could be out of service for extended periods. Technical Specifications permit plant operation for 28 days with the on-line monitor RM-A5 inoperable. During such periods, grab samples for monitoring radioactive gas every four hours would provide notice of a small increase in primary-to-secondary leakage, while other plant indications would quickly register a sudden large increase in leak rate.

We are concerned that a small increase in leak rate, which could be the precursor of a more serious problem, might go undetected for a matter of hours. The Staff considered but rejected a possible license condition that would require operability of the RM-A5 system at all times. We direct that redundancy be supplied in the form of a duplicate RM-A5 system or suitable equivalent of comparable sensitivity and response time. We further direct that the Technical Specifications be modified to permit plant operation for a maximum of 28 days with one of these duplicate systems inoperable, and to require plant shutdown if both of these systems are inoperable. As an alternative to the installation of a duplicate system, we direct that the RM-A5 system must be operable at all times during plant operation. (See Order, infra).

TMIA was concerned that leak-rate measurements might be misleading because of a tendency for some leaks to be self-sealing by build-up of corrosion products. This could occur only for very small leakage pathways between the expanded portion of a tube and the tubesheet. Accordingly, we are satisfied that this effect will not be significant from a safety standpoint. TMIA also questioned whether the loss of pretension in certain tubes might cause the measured leak rates to be reduced, potentially masking the detection of a critical size circumferential crack. Testing showed that such cracks do not exist in the tube pressure boundary. If such a crack were to appear, it would propagate only when the tube was placed in axial tension, which would tend to offset the effect of loss of pretension. We are satisfied that the loss of pretension will not be significant from a safety standpoint.

Issue 1.b Frequency of Eddy Current Tests (Fdgs. 21-25).

The Staff's proposed License Condition 3, as modified in Supplement 1 to the SER, reads as follows:

> The licensee shall conduct eddy-current examinations, consistent with the inspection plan defined in Table 3.3-1, either 90 calendar days after reaching full power, or 120 calendar days after exceeding 50% power operation whichever comes first.

The Board requested evidence on the above-captioned issue because TMIA alleged that the Staff changed its position without explanation. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

The Staff's early view was that eddy current tests (ECT) should be conducted 30-60 days after restart. This was later changed to either 90 or 120 days as reflected in the originally proposed license condition and in its modification.

Both Licensee and Staff testified that the change was justified in the light of extensive information about the condition of the steam generators that had become available in the meantime. Additional operational considerations and judgments about obtaining the maximum information from ECT were included in the decision. The Board accepts the explanations of the Licensee and Staff as sufficient rationale for the change in proposed timing of the ECT requirements.

The Board is concerned that the Staff's proposed license condition does not address the possibility of plant operation for an extended period at less than 50% power. In addition to the Staff's proposed license condition, we direct the Staff to require an assessment by the Licensee at the end of 180 days of operation at power levels between 5% and 50%, such assessment to contain recommendations and supporting information as to the necessity of a special ECT shutdown before the end of the refueling cycle. Based on this assessment, the Staff shall determine the time of the next ECT, consistent with the other provisions of the license condition. In the absence of an assessment, a special ECT shutdown shall take place before an additional 30 days of operation at power above 5%. (See Order, infra).

Issue 1.c Power Ascension Limitations (Fdgs. 26-30).

In the SER, the Staff proposed license conditions, which read as follows:

> License Condition 1. The licensee shall complete its pre-critical test program in essential conformance with the program described in its Topical Report 008, Rev. 2, and shall submit the results of that test program and a summary of its management review, prior to initial criticality.
>
> License Condition 2. The licensee shall complete its post-critical test program at each power range (0-5%, 5%-<50%, 50%-100%) in essential conformance with the program described in Topical Report 008, Rev. 2, and shall have available the results of that test program and a summary of its management review, prior to ascension from that power range and prior to normal power operation.

The Board requested evidence on the above-captioned issue because TMIA questioned whether the proposed power ascension program is in accord with the recommendations of the Third Party Review (TPR) Group.

This group was composed of consultants from the industry retained by the Licensee. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

The TPR Group recommended two hold periods at less than full power. The Licensee accepted this recommendation. The TPR Group suggested operation with one steam generator at a higher power than the other. The Licensee explained that this was not feasible, and the TPR Group accepted the explanation.

The Licensee either accepted the TPR group recommendations or provided adequate explanations. Accordingly, we find that the objections by TMIA concerning the issue of power ascension limitations are without merit.

Issue 1.d Long-Term Corrosion Tests (Fdgs. 31-43).

In the SER, the Staff proposed a license condition which reads as follows:

> License Condition 6. The licensee shall provide routine reporting of the long-term corrosion "lead tests" test results on a quarterly basis as well as more timely notification if adverse corrosion test results are discovered (SE Section 3.5).

The Board requested evidence on the above-captioned issue because TMIA asserted that accurate simulation of actual TMI-1 tube properties is virtually impossible in such tests. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

In its proposed findings, TMIA asserts that the long-term corrosion tests included a tube with a known defect but that there is no evidence with regard to the number of tube sections included in this test sequence. Although the exact number of samples was not stated, there is much evidence about the wide range of conditions simulated, and there is testimony that several samples had known defects. TMIA complains that other testing utilized archival tubes which had not been installed in a steam generator. However, this is not the case for the corrosion tests, and is relevant to a different issue (hardness tests).

TMIA asserts that Licensee has provided no assurance that tube rupture due to mechanical failure will not occur, although such assurance was outside the scope of the long-term corrosion tests and was, instead, the subject of the Licensee's and the Staff's motions for summary disposition that were granted. (Memorandum and Order, June 1, 1984). Moreover, TMIA claims that Licensee has failed to account for the mechanical stresses present in the steam generators, and complains that Licensee has not introduced transient loads into the testing sequence.

TMIA asserts that because the Licensee failed to include stresses greater than 1,100 pounds as part of the long-term corrosion program, the testing does not adequately predict operating conditions. However, the 1,100-pound loads adequately simulated heatup, operation, and cold shutdown. The tests also took into consideration residual stresses produced by the kinetic expansion process. Furthermore, C-ring specimens were loaded to a stress level slightly below yield, which is significantly higher than the level seen by tubes in actual service. Consequently, the high stress on the C-rings bounds loads induced by accident transients (a maximum of 3,140 pounds). We are therefore unable to follow the logic which TMIA uses to conclude that the maximum load that the tubes can tolerate is 1,100 pounds.

Finally, we address the complaint that Licensee did not introduce transient loads into the long-term corrosion testing sequence. It is to be noted that the issue is "adequacy of simulation of operating conditions by long-term corrosion tests," and not the simulation of all operating conditions by every conceivable type of test. Thus, the complaint is irrelevant to the matter in issue.

We conclude that the questions raised on this issue have been adequately answered, and that the Licensee's long-term corrosion test program includes a wide variety of tests which, taken together, constitutes a reasonably accurate and valid simulation of steam generator operating conditions.

Issue 2. Inadvertent Initiation of Emergency Feedwater (Fdgs. 44-47).

The Board requested that evidence be presented on this issue because neither TMIA nor the Board felt that sufficient detail was presented in the motions for summary disposition. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

TMIA did not submit proposed findings of fact on this issue, although the Board had directed the parties to file, and ruled that they would be deemed in default if they did not file, proposed findings of fact, etc. (Tr. 684). Accordingly, TMIA is deemed to be in default on this issue. Florida Power & Light Co. (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-280, 2 NRC 3, 4 n. 2 (1975).

However, the issue is addressed in this opinion because the Board had expressed an uncertainty about the maximum transient stresses associated with inadvertent initiation of emergency feedwater. Our uncertainty has been resolved by the explanation that the high heat transfer rate from steam to subcooled water would cause the incoming water to be heated sufficiently that its effect on tube loads would be insignificant. Further, with respect to rapid cooldown following a loss-of-coolant accident, emergency feedwater injection was already included in calculating the maximum stress.

Issue 3. Hardness Tests on Repaired Tubes (Fdgs. 48-60).

The Board requested that evidence be presented on this issue because the absence of post-repair hardness tests on corroded tubes was not sufficiently explained by the Licensee. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

We are satisfied that hardness testing of repaired tubes in place is not feasible because of the size of the measuring equipment. We are also satisfied that removal of samples is impractical because of radiation exposure to workers.

Hardness testing was performed on archival tube samples that had undergone kinetic expansion. It was demonstrated in a reasonable number of tests that archival and actual tubes had the same mechanical properties, especially with regard to the key parameters of ductility and yield strength which are used to judge suitability for kinetic expansion. Hardness tests on the kinetically expanded archival tubing indicated less residual stress in the transition region than in the original as-fabricated rolled tubes.

In its proposed findings, TMIA asserted that the purpose of hardness tests is to determine the degree of embrittlement, and that embrittlement dictates loss of ductility and yield strength. Actually, hardness tests were conducted to assess the degree of "cold working" and the susceptibility to intergranular stress-assisted cracking (IGSAC). The small increase in hardness introduced by the expansion process produces essentially no change in ductility.

TMIA asserted that no meaningful conclusions can be drawn from a comparison of the results of tests conducted on different populations of tubing. While this may be true as a general principle, the three tests in question involved a prudent selection of archival tube samples. Other tests were performed on actual TMI-1 tubes.

TMIA tried to make an issue of the use of the words "identical" and "representative" in comparing various tube samples. It appears that responses to Board questions on different topics were taken out of context, and that no issue exists.

In response to other objections raised by TMIA, we find that the number of samples of actual tubes used for yield stress measurements was reasonable, and we find no safety significance in statistical variations among pull-out load tests using test samples under different test conditions.

We note that a number of these matters concern Licensee's qualification testing program, which was ruled to be outside the scope of Contention 1.a (Memorandum and Order, June 1, 1984, at 14). These matters are addressed here only because the Board asked some supplemental questions about how closely the archival tubes corresponded to the actual tubes in TMI-1. (Tr. 526-52).

In our opinion, the Licensee has presented reasonable justification for not performing post-repair hardness testing on kinetically-expanded TMI-1 steam generator tubes.

Issue 4. Industry Experience with Kinetic Expansion (Fdgs. 61-65).

The Board requested that evidence be presented on this issue solely because the Licensee's motion for summary disposition asserted that the use of kinetic expansions to seal heat exchanger tubes within tubesheets has a broad base of successful experience. Licensee did not state whether this experience includes nuclear plant components, or whether the experience includes repair of damaged heat exchangers, manufacture of new heat exchangers, or both. Information was requested about whether tube integrity during subsequent operation depends on whether the process is a repair, or a manufacturing process using new materials. The Licensee presented a witness from Foster Wheeler Development Corporation. The Staff presented a panel of witnesses.

There is no evidence before us that the kinetic expansion process has been used for repairing steam generator tubes in nuclear power plants. The industry has had considerable experience with this process in other types of heat exchangers, both in field repairs and in fabrication. This experience indicates that the integrity of kinetically expanded joints depends primarily on key parameters (yield strength and ductility) irrespective of whether the process is applied to new equipment during fabrication or the repair of existing equipment.

However, the extensive repairs to the TMI-1 steam generators is a new, large-scale application of the kinetic expansion process. Because there is no directly relevant experience, approval of these repairs must be based on the other issues discussed in this opinion. Accordingly, we conclude that this issue has little significance in the resolution of this contention.

B. CONTENTION 1.b (Fdgs. 66-75).

TMIA's Contention 1.b, as originally admitted, alleged as follows:

> Because of the enormous number of tubes in both steam generators which have undergone this repair process, (1) the possibility of a simultaneous rupture in each steam generator, which would force the operator to accomplish cooldown and depressurization using at least one faulted steam generator, resulting in release of radiation into the environment beyond permissible levels, "isn't an incredible event," (see, September 19, 1982 memorandum from Paul Shewmon, then Chairman of the ACRS), (2) and could lead to a sequence of events not encompassed by emergency procedures, (3) and in the course of a LOCA, such a scenario could create essentially uncoolable conditions.

As noted in Part IA, above, in the unpublished Memorandum and Order of June 1, 1984, at page 32, the Board denied in part the Licensee's and the Staff's motions for summary disposition of Contention 1.b, and requested that evidence be presented at the hearing as to whether the kinetic expansion tube repair process increased the probability of simultaneous tube ruptures in both steam generators. The Board requested evidence on this one issue because it had been raised in an Advisory Committee on Reactor Safeguards (ACRS) memorandum concerning

TMI-1 and because the Board wanted more information on which to base its decision. The Licensee and the Staff each presented a panel of witnesses to testify on this issue.

A steam-generator tube rupture, as it is commonly understood in the industry, cannot take place at or in the vicinity of the repair joint. A break producing a large flow has no room to occur because the new joint is closely confined within the tubesheet hole. Moreover, the evidence justifies a conclusion that the repair did not significantly affect tube strength and ductility, so that the probability of tube ruptures has not been increased by the repair.

In its proposed findings, TMIA raises an issue for the first time, namely, that tube rupture caused by rubbing and wearing of adjacent bowed tubes could occur when compressive loads are applied to adjacent tubes that had lost preload. This seems very unlikely to cause a problem because contact between such tubes, even if possible, would not occur during steady operation, but only during heatup, which lasts about 8-10 hours.

TMIA would also have us rule that Licensee has not provided adequate assurance that the repair has significantly reduced the probability of simultaneous tube rupture. This is not the issue. The central issue is whether the repair process has increased the probability of such an accident. We find that reasonable assurance exists that the repair process has not increased the probability of simultaneous tube ruptures in both of TMI-1's steam generators.

III. CONCLUSIONS

The Board concludes that the license conditions proposed by the Staff, as supplemented by the Board's two imposed conditions as discussed in Section II, supra, and the Licensee's post repair and plant performance testing and analysis provide reasonable assurance that the leak tight integrity of the repaired steam generator tube joints will be maintained. We further conclude that the uncertainties which led the Board to request the presentation of evidence on specific issues have been resolved, and that reasonable assurance exists that the repair process has not increased the probability of simultaneous tube ruptures.

FINDINGS OF FACT

A. CONTENTION 1.a

TMIA Contention 1.a asserts the following:

> Neither Licensee nor the NRC Staff has demonstrated that the kinetic expansion steam generator tube repair technique, combined with selective tube plugging, provides reasonable assurance that the operation of TMI-1 with the as-repaired steam generator can be conducted without endangering the health and safety of the public, for the following reasons:
>
> a. Post repair and plant performance testing and analysis including the techniques used, empirical information collected, and data evaluation, and proposed license conditions are inadequate to provide sufficient assurance that tube ruptures, including but not limited to those which could result upon restart, a turbine trip at maximum power, thermal shock from inadvertent actuation of emergency feedwater at high power or following rapid cooldown after a LOCA, will be detected in time and prevented to avoid endangering the health and safety of the public through release of radiation into the environment beyond permissible limits.

1. The Board's Memorandum and Order of June 1, 1984 (unpublished), in partially denying the Licensee's and Staff's motions for summary disposition of TMIA's Contention 1.a, identified the following issues as to which evidence should be presented at the hearing:

> 1. The rationale underlying certain proposed license conditions should be addressed, with attention to:
>
> a. Reliability of leak rate measurements.
>
> b. Method of determining frequency of ECT tests.
>
> c. Method of determining power ascension limitations.
>
> d. Adequacy of simulation of operating conditions by long-term corrosion tests.

2. The effect of inadvertent initiation of emergency feedwater flow at high power or following rapid cooldown after a LOCA should be addressed, with attention to calculation of maximum transient stresses in steam generator tubes.

3. The reasons for not including hardness tests on repaired tubes in the post repair testing program should be addressed.

4. Recalling Licensee's statement in ¶¶ 1-8 [of its Statement of Material Facts As To Which There Is No Genuine Issue To Be Heard] that the use of kinetic expansions to seal heat exchanger tubes within tubesheets has a broad base of successful experience, information is requested about whether tube integrity during subsequent operation depends on whether the process is a repair, or a manufacturing process using new materials.

(Our findings of fact with regard to Contention 1.a are captioned according to the preceding list of issues.)

Issue 1.a Reliability of Leak Rate Measurements.

2. Primary-to-secondary leak rate measurements during PWR operation are made to document the absolute value of leakage and to document any trends which may be cause for concern. The absolute value is required to assess performance of steam generators and to ensure that technical specification limits are not exceeded. Trends are monitored because increasing leakage may indicate ongoing chemical or mechanical degradation. Increasing leak rates are investigated further to identify leak locations and take appropriate corrective action. (Licensee's Test., fol. Tr. 224, at 5-6).

3. Technical Specifications 3.1.6.3 and 4.1 address primary-to-secondary leakage through TMI-1 once-through steam generator (OTSG) tubes. Technical Specification 3.1.6.3 requires that if this leakage exceeds 1 GPM total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours. Technical Specification 4.1

requires that leakage be evaluated daily. (Licensee's Test., fol. Tr. 224, at 3).

4. The Staff's proposed License Condition 4, as modified in Supplement 1 to the SER, reads as follows:

> The licensee shall confirm the baseline primary-to-secondary leakage rate established during the steam generator hot test program. If leakage exceeds the baseline leakage rate by more than 0.1 GPM, the plant shall be shut down and leak tested. If any increased leakage above baseline is due to defects in the tube free span, the leaking tube(s) shall be removed from service. The baseline leakage shall be re-established, provided that the present Technical Specification limit of 1.0 GPM is not exceeded (SE Section 3.3).

(Board Exhibit 2, p. 27).

5. Licensee determined the baseline primary-to-secondary leakage to be 0.02 GPM (1 GPH) during the steam generator hot test program. This means that the facility is to be shut down if the leak rate reaches 7 GPH total for both steam generators, as compared to the existing limit of 60 GPH in Technical Specification 3.1.6.3. (Licensee's Test., fol. Tr. 224, at 4). Because of recently discovered leakage and associated repairs, the baseline leakage rate will be re-established on restart of the plant. (Tr. 327).

6. The TMI-1 leakage limitations in Technical Specification 3.1.6.3 are comparable to those at most other pressurized water reactors in the United States. The proposed new limit of 0.1 GPM is the most restrictive limit implemented at any plant. (Licensee's Test., fol. Tr. 224, at 5; Tr. 240; Staff's Test., fol. Tr. 589, at 8; Tr. 611).

7. The steam generator hot testing results indicate that a monitored leak rate statistical variation (twice the standard deviation

from the mean value) of approximately ±0.01 GPM (±0.5 GPH) can be expected during steady state operation. (Licensee's Test., fol. Tr. 224, at 7).

8. Primary-to-secondary leakage is indicated by several diverse methods at TMI-1. These methods include measuring radioactive noble gas concentrations on the secondary side, and measuring chemistry and radio-chemistry in secondary side OTSG water. The radionoble gas concentration measurement is the most sensitive method of quantifying the primary-to-secondary leakage rate. The leakage rate is calculated periodically by utilizing data from on-line continuous monitors and grab sample analysis. (Licensee's Test., fol. Tr. 224, at 8).

9. Continuous leak rate monitoring during operation is provided by a flow rate instrument and by a radiation detector. The radiation detector is monitored in the control room and is alarmed. (Tr. 240-41). The response time for the radiation detection system RM-A5L is of the order of a few minutes. (Tr. 274-75). The sensitivity of this system is at least 0.001 GPM (0.07 GPH) during power operation and 0.003 GPM (0.2 GPH) during plant cooldown. (Licensee's Test., fol. Tr. 224, at 10). There are additional radiation monitors which are less sensitive than RM-A5L. At least one of the other monitors would come on scale before the leak rate would reach the Technical Specification Limit. (Tr. 267-68).

10. Regular measurements of radioactivity in grab samples of condenser off-gas provide leak rate information even if the on-line

monitors are not functioning. (Tr. 268). Ordinarily, these samples are taken every eight hours. (Tr. 624). The plant could conceivably operate at full power for as much as eight hours without detection of a small increase in leak rate. (Tr. 642). Technical Specifications permit plant operation for 28 days with the on-line monitor RM-A5 inoperable, provided grab samples are being taken. (Tr. 646). Licensee has an administrative limitation that if RM-A5 is determined to be out of service, a grab sample will be taken immediately and repeated every four hours. (Tr. 647). The Staff considered but rejected a possible license condition that would require operability of the RM-A5 system at all times. (Tr. 643).

11. Two cold leak tests are used, the bubble test and the drip test. The bubble test is the most sensitive cold leak test, having a leak rate sensitivity of 0.000005 GPM for an individual leak. The bubble test was used on the upper portions of the OTSG tubes which included all the new kinetic expansion joints. (Licensee's Test., fol. Tr. 224, at 10-11).

12. The entire OTSG tube length is leak tested by the drip test. Sensitivity depends on location, being 0.0002 GPM near the lower tubesheet and 0.002 GPM at the high ends of the tubes. (Licensee's Test., fol. Tr. 224, at 11).

13. Leak rate measurements determine total primary-to-secondary leakage, including the contribution from leakage through the joints. (Licensee's Test., fol. Tr. 224, at 12). Some leakage is to be expected, and small leakage through joints does not relate to their

load-carrying capability. (Tr. 269). If the observed leak rate should increase by 0.1 GPM, the plant will be shut down and the individual leaking tubes, plugs and/or joints will be identified by means of the bubble and drip tests. (Licensee's Test., fol. Tr. 224, at 12).

14. There may be a tendency for some leaks to be self-sealing, but only for leakage pathways between the expanded portion of the tube and the tubesheet. The joint is formed between the inconel tube and the carbon steel tubesheet. Corrosion products tend to plug up leakage paths in the tight tube-to-tubesheet crevice and to stop or slow leakage. This was confirmed by a trend of decreasing leakage with time for joints tested in Licensee's qualification program. (Licensee's Test., fol. Tr. 224, at 12; Tr. 245-46; Tr. 271-72).

15. To be self-sealing, a leak past the joint would have to have a very small flow through a pathway sufficiently tight to enable the build-up of corrosion products adequate to seal the leak. A leak of this size would not adversely affect the load bearing capability of the joint, or increase the probability of rupture within the joint. (Licensee's Test., fol. Tr. 224, at 12-13; Tr. 269; Tr. 260-64).

16. Leakage past a repaired joint is independent of the loss of pretension. Pretension, or preload, was placed on the tubes by thermal expansion during the manufacture of the steam generators. At TMI-1, some tubes with complete circumferential cracks were freed from the original joints. These tubes contracted slightly, relieving all or part of the pretension. After kinetic expansion, these tubes were again fixed at both ends, but without some or all of the original pretension.

This resulted in a reduction of axial tube load of several hundred pounds. (Licensee's Test., fol. Tr. 224, at 13; Tr. 257-58).

17. The kinetic expansion process relies on horizontal (radial) forces to expand tubes, while pretension is an axial (vertical) load. These load components are perpendicular to each other, and the loss of pretension does not affect the ability to expand the tube and form the new joint. Kinetically expanded joints in tubes with loss of pretension are as tight, and no more prone to leakage, than joints in tubes with preload. Monitoring of leakage through such a joint is unaffected by a loss of pretension. (Licensee's Test., fol. Tr. 224, at 13-14).

18. A tube without pretension might theoretically exhibit a lower leak rate than a tube with pretension for a circumferential through-wall crack of a given size, hence potentially masking the detection of a critical size crack. Testing already conducted shows that such cracks do not exist in the tube pressure boundary. If such a crack were to exist, it would propagate due to intergranular stress-assisted cracking (IGSAC) only when the tube was placed in axial tension, which would tend to offset the effect of loss of pretension. (Licensee's Test., fol. Tr. 224, at 14; Tr. 273).

19. During the steam generator hot testing program, transients placed axial tensile loads of at least several hundred pounds on every tube in the steam generators, including those which had lost preload. Measured leak rates were compared with calculations. The results confirmed the conclusion that no large cracks remain undetected. (Licensee's Test., fol. Tr. 224, at 15; Tr. 276-83).

20. If future cracks were to form and propagate due to IGSAC at normal operating conditions, the principal direction of propagation will be axial (along the tube). IGSAC propagation is perpendicular to the direction of highest stress, which is circumferential (hoop stress) at normal operating conditions. Therefore, a loss of pretension would not affect measurements of leakage from axial tube cracks. (Licensee's Test., fol. Tr. 224, at 15).

Issue 1.b Frequency of Eddy Current Tests.

21. The Staff's proposed License Condition 3, as modified in Supplement 1 to the SER, reads as follows:

> The licensee shall conduct eddy-current examinations, consistent with the inspection plan defined in Table 3.3-1, either 90 calendar days after reaching full power, or 120 calendar days after exceeding 50% power operation whichever comes first. (SE Section 3.3).

(Board's Exhibit 2, p. 27).

22. In recommending the change in eddy current test (ECT) frequency, which the Staff subsequently incorporated into the proposed license condition, the Licensee considered the condition of the generator, the type of repairs, the damage mechanism leading to the repairs, and the expectation that if any new damage were to occur, it would be at a slow rate. Additional considerations were plant accessibility, other operational sequences being conducted, and prudent operating practices which dictate that the opening of steam generators, with its attendant exposure to oxygen, should be minimized. (Licensee's Test., fol. Tr. 226, at 4-5).

23. Since the ECT program is designed to characterize change, there is a need to allow reasonable operating time on the generators to allow any unforeseen mechanism to cause change. The full benefits of ECT can only be obtained after operation at some period of time to allow the system to approach chemical equilibrium. (Licensee's Test., fol. Tr. 226, at 7-8).

24. The Staff recommended in May 1982 that the plant be operated for 30 to 60 days and then be shut down for eddy current tests to assess the progression of degradation. The Staff subsequently changed its position because a large amount of information on the rate of progression, the type of attacks, the corrosive species, etc., became available. (Tr. 606).

25. The proposed license condition does not contain a requirement for a special shutdown for ECT in the event that the plant is operated for an extended period at less than 50% power. The Staff witness had not anticipated this possibility, but stated that if it were to occur he would be inclined, after approximately 180 or 200 days, to tell the Licensee that the Staff would like them to shut down and conduct eddy current tests, which he assumes they would be willing to do. (Tr. 672-73).

Issue 1.c Power Ascension Limitations

26. The subject of power ascension limitations is addressed in the Staff's proposed license conditions in the SER which read as follows:

> License Condition 1: The licensee shall complete its precritical test program in essential conformance with the program described in its Topical Report 008, Rev. 2, and shall submit the results of

that test program and a summary of its management review, prior to initial criticality.

License Condition 2: The licensee shall complete its postcritical test program at each power range (0-5%, 5%-<50%, 50%-100%) in essential conformance with the program described in Topical Report 008, Rev. 2, and shall have available the results of that test program and a summary of its management review, prior to ascension from that power range and prior to normal power operation.

(Board's Exhibit 1, p. 46).

27. The initial power ascension program was developed prior to knowledge of the damage to the steam generators. In conjunction with the steam generator repair program, special pre-critical tests were developed to demonstrate steam generator operability. These tests have now been performed and evaluated. It was determined, however, that two 30-day hold periods should be added to the power ascension program. (Licensee's Test., fol. Tr. 229, at 4-5).

28. Proposed license conditions Nos. 1 and 2 are not intended to limit power ascension. Rather, they are intended to require that test results be made available to the Staff at each stage of the test program. (Staff's Test., fol. Tr. 589, at 10; Tr. 639-40).

29. In its report of February 18, 1983, the Third Party Review (TPR) Group, which was composed of consultants from industry retained by the Licensee, recommended consideration of a "substantially extended operation at low power" and of a "hold period of perhaps a month or more at 40 percent power . . . followed by another month or more at 70 percent power . . . ." In accordance with the TPR recommendation, Licensee modified the power ascension program to add two 30-day hold periods, one at 48% power and one at 75%. In its report of May 16,

1983, the TPR Group stated that the GPU Nuclear response is satisfactory. (Licensee's Test., fol. Tr. 229, at 6-7; Staff Exhibit 1, at 3).

30. The TPR Group also recommended that Licensee "consider the possibility of deliberately running one steam generator at a higher power than the other during the first escalation hold periods." Licensee explained to the TPR Group that this suggestion was not feasible; in particular, it would involve mismatched reactor coolant flow, imbalanced feed flows, and different coolant levels in each generator. This could mask changes in plant conditions, including any abnormalities in the plant response to transients. In its report of May 16, 1983, the TPR Group stated that the GPU Nuclear response is satisfactory. (Licensee's Test., fol. Tr. 229, at 7-8).

Issue 1.d Long-Term Corrosion Tests.

31. Long-term corrosion tests are required in a license condition proposed by the Staff in the SER which reads as follows:

> License Condition 6: The licensee shall provide routine reporting of the long-term corrosion "lead tests" test results on a quarterly basis as well as more timely notification if adverse corrosion test results are discovered (SE Section 3.5).

(Board Exhibit 1, p. 46).

32. The purpose of the long-term corrosion test program, the operations phase of which has now been completed, is to verify that sulfur-induced intergranular stress-assisted cracking (IGSAC) will not reinitiate or propagate in the TMI-1 steam generators under actual operating conditions. The tests were designed to confirm that

metallurgical, environmental, geometric and surface conditions which exist after the repair of the tubes are not detrimental to tube integrity. From the test program it will be possible to conclude whether the proposed chemistry limits are acceptable, whether the peroxide cleaning was beneficial or damaging, and whether the changes in electrochemical potential during operations will cause reinitiation of corrosion. (Licensee's Test., fol. Tr. 231, at 2-3).

33. The long-term corrosion tests are accordingly related to the kinetic expansion repair process, but only insofar as they verify that the repair did not render the steam generators susceptible to reinitiation of IGSAC. This is tested by including kinetically expanded tube samples in the test loops. Except in this one respect, the long-term corrosion tests have no relationship to the adequacy of the kinetically expanded joint. The tests were not designed to confirm assurance against the possibility of mechanically induced tube ruptures caused by various transients, and the tests provide no information on this subject. (Licensee's Test., fol. Tr. 231, at 3).

34. The long-term corrosion test program includes tests which closely simulate the typical operating environment of the steam generator tubing during steady state and transient conditions. The program also includes comparative tests which closely simulate steam generator operation, but using tubes with high residual sulfur levels (not peroxide cleaned) exposed to thiosulfate. The tests reproduced all the parameters which influence IGSAC, i.e., susceptible material, environment, and stress. (Licensee's Test., fol. Tr. 231, at 4-5).

35. To assure that the influence of prior operation and layup on tubing was adequately represented, only tube sections removed from the TMI-1 steam generators were used as specimens. These specimens were selected from various regions of each steam generator, including tube sections which had known defects. (Licensee's Test., fol. Tr. 231, at 5; Staff's Test., fol. Tr. 589, at 12).

36. The tube sections for the long-term corrosion tests were selected from tubes that had been previously removed from the steam generators for use in the failure analysis. Sections were selected to provide a maximum range of properties. Tests specimens were selected from representative heats of material removed from the generator in order to provide a range of typical chemistry. Yield strengths of the specimens spanned the range of tubes in the steam generators. Specimens were selected that displayed various levels of susceptibility to corrosion damage; some came from tubes with no defects and others from tubes with as many as eight indications. (Licensee's Test., fol. Tr. 231, at 5-6; Staff's Test. fol. Tr. 589, at 11-12; Tr. 353-55).

37. The test samples were representative of tubes from various axial locations in each steam generator. The samples were also representative of various heats, and bounded the heats of the metal in the tubes. No correlation could be found between heat number and any propensity for cracking. (Licensee's Test., fol. Tr. 231, at 6; Tr. 353-56).

38. Certain samples were subjected to the explosive expansion process using mockup tubesheets and then subjected to peroxide cleaning.

Other samples were not peroxide cleaned, in order to test what could occur if the cleaning process had not been undertaken. Some C-ring samples made from actual TMI-1 tubes were also included. These samples provided a means for metallographic examination to observe any microstructural changes or incipient cracking. (Licensee's Test., fol. Tr. 231, at 6).

39. Environmental chemistry parameters were selected to either simulate, or be more aggressive than, the water chemistry which will be maintained in the reactor coolant system. Three of the four test loops had 100 ppb of sulfate, the maximum permitted under operating chemistry specifications. The fourth loop had 100 ppb of thiosulfate. Maximum permitted levels of chloride and fluoride were also used. (Licensee's Test., fol. Tr. 231, at 6-7).

40. The tests included typical temperature cycles. Temperatures were raised from ambient to normal operating temperature (approximately 600°F). Temperatures were held constant at operating level, and also cycled between 500°F and 600°F to simulate unit load changes. The test loops were also subjected to cooldown cycles, some of which included the introduction of oxygen. Tests also simulated the atmospheric and temperature conditions present at the time of the original IGSAC damage. (Licensee's Test., fol. Tr. 231, at 7).

41. During heatup, operation, and cooldown, tubes in the steam generators undergo changes in stress. A net axial tensile stress could exist during cold shutdown and steady state operation. The stress is reduced during heatup and increased during cooldown due to differential

thermal expansion effects. In order to simulate the changes in axial load, full tube specimens were loaded at a level corresponding to steady state loads during heatup, cold shutdown, and operation. During cooldown, loads were increased to approximate the maximum allowed cooldown rate. Full tube specimens simulating repaired joints were kinetically expanded to ensure representative residual stresses and then exposed to the axial loads described above. (Licensee's Test., fol. Tr. 231, at 8; Tr. 359).

42. The C-ring specimens were loaded to a level just slightly below yield, which is significantly higher than the load seen by tubes in actual service. This would make them more susceptible to IGSAC than are the actual steam generator tubes. This also bounds any load that would be experienced under accident transients. (Licensee's Test., fol. Tr. 231, at 8; Tr. 369-70; Tr. 540-42).

43. The tests were not designed to simulate fatigue damage. Results of the tests simulating heatup and cooldown cycles were sufficient to predict the effect of stress on corrosion. (Tr. 345-46).

Issue 2. Inadvertent Initiation of Emergency Feedwater.

44. In the unlikely event of inadvertent actuation of the emergency feedwater (EFW) pumps in conjunction with inadvertent opening of the EFW valves, resulting in injection of emergency feedwater into steam generators at full power, the resulting thermally induced axial tube load would not be sufficient to cause rupture of steam generator tubes. (Licensee's Test., fol. Tr. 421, at 4; Staff's Test. fol. Tr. 589, at 13-14).

45. Emergency feedwater is injected horizontally into the steam generator tube bundle steam space via six auxiliary feedwater nozzles located at approximately equal spacing around the circumference of the steam generator shell. The injection points are near the top of the tube bundle, with nozzle centerlines 2'11" below the bottom surface of the upper tubesheet. (Licensee's Test., fol. Tr. 421, at 4).

46. As the EFW is injected into the steam space, the high heat transfer rate from the steam quickly heats the incoming water. By the time the EFW reaches the tubes, it is approaching the same temperature as the steam. The affected tubes experience only a small temperature change in the small portion of the tube being sprayed, which results in an insignificant axial load change in the tube. (Licensee's Test., fol. Tr. 421, at 4-5).

47. Conservative calculations, which do not take into account the high heat transfer rate from steam to subcooled water, predict that the maximum change in load of the affected tubes would be 70 pounds tension. This would be comparable to loads at full power and much less than the loads in cooldown or design basis accident conditions. The maximum transient loads on the tubes following a loss-of-coolant accident (LOCA) have been conservatively calculated to be 2,641 pounds, including the effect of EFW injection. This load is less than the design basis load of 3,140 pounds. (Licensee's Test., fol. Tr. 421, at 5-6; Tr. 433-35; Tr. 439-40).

Issue 3. Hardness Tests on Repaired Tubes.

48. Hardness is a metallurgical term which defines the resistance of metals or alloys to plastic deformation, usually by indentation. Sometimes it also refers to resistance to scratching, abrasion or cutting. (Staff's Test., fol. Tr. 589, at 16).

49. Hardness of a metal or alloy increases when the material is subjected to "cold working" as in mechanical deformation. This can result in higher residual tensile stress, which can be indicative of increased susceptibility to intergranular stress assisted cracking (IGSAC). (Licensee's Test., fol. Tr. 423, at 3).

50. The kinetic expansion process resulted in cold working of the expanded portions of the tubes, which increased the hardness of the material. The roll expansion process used in the original tube-to-tubesheet joint also produced cold working and increased the material's hardness. (Licensee's Test., fol. Tr. 423, at 3).

51. Hardness testing during the qualification program showed the kinetically expanded joints to be less hard, and therefore to have less cold working, than non-stress relieved rolled joints. Less cold working results in lower residual stresses. This suggests that the kinetically expanded joints will be less susceptible to IGSAC than are non-stress-relieved rolled joints. (Licensee's Test., fol. Tr. 423, at 4; Staff's Test., fol. Tr. 589, at 17).

52. Hardness was not considered a parameter indicative of the adequacy of the kinetic expansion joint. The joint was qualified for a range of material tensile strengths bracketing those of the TMI-1 steam

generator tubes and a range of possible tubesheet annulus geometries and conditions. (Licensee's Test., fol. Tr. 423, at 4).

53. Hardness tests were performed on TMI-1 archival tubes which had been kinetically expanded in the same manner as the actual tubes in the steam generator. Archival tubes are tubes which were set aside as a matter of record from the same manufacturing lot or heat as those used in the steam generators. (Tr. 384; 441-42, 464-65).

54. In using archival tubes in the qualification program, including hardness testing, Licensee selected heats of archival tubes which bracketed the range of properties of heats present in the steam generators. Licensee also tested tubing removed from the steam generators to determine that the relevant properties were unchanged such that valid and representative conclusions could be drawn from tests conducted on archival tubing. The tensile strength and ductility were determined quantitatively for TMI-1 tube specimens of varied heats, and compared with pre-operational mill specification testing results for the same heats of materials. The specimens which had been in operation at TMI-1 performed within the range of expected behavior for the heat as manufactured. (Tr. 461-64; 514-15; 527; 546-48). Strip specimens bent around mandrels exhibited the high ductility expected for Inconel-600 and showed no incipient damage. (Tr. 515; 572-73). An actual TMI-1 tube specimen containing a crack was kinetically expanded, and the crack did not grow. (Tr. 472-75; 515-16). Retention of yield strength and ductility is expected behavior for Inconel-600 in steam generators. (Tr. 528-48; 634-35).

55. Hardness testing is done with relatively large equipment, and cannot be performed on the repaired tubes within the steam generator. To measure hardness, tubes would have to be severed, sectioned, and removed from the repaired steam generators. This is an extensive effort which would result in radiation exposure to workers. (Licensee's Test., fol. Tr. 423, at 4; Staff's Test., fol. Tr. 589, at 17).

56. Inconel-600 tubing maintains its mechanical strength and ductility even after extended service in steam generators, and the material does not become embrittled. Sensitization does not significantly alter the mechanical strength or ductility. (Staff's Test., fol. Tr. 652, at 2-4; Tr. 655-56).

57. Hardness tests were done on samples within the transition region and the fully expanded region of a kinetic expansion, a rolled expansion, and an unexpanded tube. Archival tubes were used for these tests. Other hardness tests were performed on actual TMI-1 tubes. (Tr. 441-42; 542-43).

58. Licensee's witness agreed that archival and actual tube samples were "identical" as far as one could tell in testing. (Tr. 465). Another witness for Licensee later used the word "identical" in describing samples removed from the steam generators that were characterized in Board questions as "typical" or "representative" of corroded tubes. (Tr. 531).

59. Of the 27 tubes removed from the steam generators for testing, three heats and at least three tubes were tested for yield stress.

(Tr. 572). These are representative of the tubes remaining in the steam generators. (Tr. 668-69).

60. Statistically significant differences among results of pull-out load tests were explained as resulting from differences in test conditions, and do not indicate that the tests failed to meet their objectives. (Tr. 567-70).

Issue 4. Industry Experience with Kinetic Expansion.

61. The kinetic expansion seal is an effective means of sealing heat exchanger tubes within tubesheets, whether performed as a field repair or as part of the original fabrication. The industry has had considerable experience with this process in both situations. (Licensee's Test., fol. Tr. 379, at 2).

62. For a power station (nuclear or fossil), there are different kinds of heat exchangers (e.g., feedwater heaters, moisture separator reheaters, etc.), most of which are of the shell and tube type. A TMI-1 steam generator is one type of shell and tube heat exchanger. (Licensee's Test., fol. Tr. 379, at 3).

63. Foster Wheeler initially used kinetic expansion to support its shop fabrication. Foster Wheeler has expanded some five million tubes. Since 1967, Foster Wheeler has adopted kinetic expansion as the primary means of tube expansion for high pressure feedwater heaters. Since the mid-seventies, Foster Wheeler has also applied kinetic expansion routinely to field repairs, including expansions similar to what was done on the TMI-1 steam generators. (Licensee's Test., fol. Tr. 379, at 3-4).

64. Other manufacturers have used kinetic expansion. (Tr. 490; 511; 620; 630). Kinetic expansion has been used in Japan, both in manufacturing and as a means of closing crevices. (Tr. 630-32).

65. The integrity of kinetically expanded joints depends primarily on key parameters (yield strength and ductility), irrespective of whether the process is applied to new equipment during fabrication or to the repair of existing equipment. (Licensee's Test., fol. Tr. 379, at 5).

B. CONTENTION 1.b

TMIA Contention 1.b asserts the following:

> Neither Licensee nor the NRC Staff has demonstrated that the kinetic expansion steam generator tube repair technique, combined with selective tube plugging, provides reasonable assurance that the operation of TMI-1 with the as-repaired steam generator can be conducted without endangering the health and safety of the public, for the following reasons:
>
> * * * * *
>
> b. Because of the enormous number of tubes in both steam generators which have undergone this repair process, (1) the possibility of a simultaneous rupture in each steam generator, which would force the operator to accomplish cooldown and depressurization using at least one faulted steam generator, resulting in release of radiation into the environment beyond permissible levels, "isn't an incredible event," (see, September 19, 1982 memoranoum from Paul Shewmon, then Chairman of the ACRS), (2) and could lead to a sequence of events not encompassed by emergency procedures, (3) and in the course of a LOCA, such a scenario could create essentially uncoolable conditions.

66. The Board's Memorandum and Order of June 1, 1984 (unpublished), in partially denying the Licensee's and the Staff's motions for summary disposition of TMIA's Contention 1.b, requested that evidence be presented at the hearing upon whether the kinetic expansion

tube repair process increased the probability of simultaneous tube ruptures in both steam generators.

67. A steam-generator tube rupture, as it is commonly understood in the industry, cannot take place at or in the vicinity of the repair joint. A break producing a large flow has no room to occur because the new joint is closely confined within the tubesheet hole. Moreover, any leakage would be significantly restricted by the tight crevice formed by the tubesheet hole and the outside of the tube. (Tr. 508-10; 476-77).

68. Inconel retains its strength and ductility despite previous operation of the steam generators. Test results indicate that the repair itself did not affect strength and ductility. The probability of simultaneous tube ruptures involving both steam generators is not significantly greater now than it was at the time of the original licensing. (Staff's Test., fol. Tr. 652, at 5).

69. The design basis transients specified for the original design tube-to-tubesheet joint were specified as applicable to the repaired steam generator tube-to-tubesheet joint. The repair joint was qualified by testing and analysis for transients in a postulated main steam line break load of 3,140 pounds tension, the maximum design basis loading of the tube-to-tubesheet joint. All other normal operating or postulated accident loadings are enveloped by this loading. Moreover, the only conceivable failure of the kinetic expansion joint would be by slippage under applied axial load, rather than by tube rupture. (Licensee's Test., fol. Tr. 425, at 3-4; Tr. 509-10).

70. The kinetic expansion repair produces a new transition zone between the expanded and non-expanded portions of the tube. A similar transition zone existed previously at the original roll expansion. However, the transition for the kinetic expansion was carefully developed to be more gradual than that of the original shop roll expansion, and, in general, the kinetic expansion process tends to result in less cold working than the roll expansion process. While the residual stresses in the kinetic expansion transition may be slightly higher than those in roll expansions which have experienced the fabrication stress relief heat treatment, residual stresses and the amount of cold working in the kinetic expansion are both less than in non-stress-relieved roll expansion transitions for which there is a considerable body of satisfactory operating experience in nuclear power plants. (Licensee's Test., fol. Tr. 425, at 5; Tr. 410-13; Tr. 465-68; Tr. 489-97; Tr. 506; Tr. 634).

71. The residual stresses within the transition zone are not a concern from a static or fatigue stress standpoint, but could affect the susceptibility of the material to intergranular stress-assisted cracking (IGSAC). The resistance of the kinetic expansion transition zone to IGSAC is demonstrated by operating experience of once-through steam generators containing non-stress-relieved roll expansions, and by Licensee's accelerated and long-term corrosion testing. (Licensee's Test., fol. Tr. 425, at 5; Tr. 497).

72. To date, there have been no failures, by cracking in the transition zone, of tubes with non-stress-relieved roll expansions in

B&W once-through steam generators in service. Short-term (accelerated) corrosion testing, which was performed as part of the TMI-1 qualification program, showed no evidence of cracking in either kinetic or non-stress-relieved roll expansion transitions during the simulated life of the repair when exposed to a caustic (10% NaOH at constant potential) environment. Thus, the likelihood of tube rupture of the new transition due either to loading or IGSAC is no greater than that for tubes currently operating in other once-through steam generators. (Licensee's Test., fol. Tr. 425, at 5-6).

73. The potential effects of the kinetic expansion process on the balance of the tube were also carefully evaluated. The only effect warranting further analysis was the change in tube preload. The kinetic expansion repair process produces less than a 30-pound decrease in tube preload for normal steam generator tubes. A small percentage of the tubes in the steam generators may have lost all preload due to the IGSAC completely severing the tube in or near the original roll expansion at the top of the tube. This allowed the tube to slip down slightly and relieve the existing preload in the tube. In some cases, vibrations from nearby kinetic expansions may have contributed to the slipping process. (Licensee's Test., fol. Tr. 425, at 6; Tr. 477-78).

74. The increase in the compressive load due to loss of any or all of the tube preload when added to the maximum compressive load (which occurs during a normal heat-up transient of 100°F/hr) is less than the compressive load required to cause contact between adjacent tubes. Accordingly, there is no increased potential for tube ruptures due to

increased wear. Furthermore, the loss of the tube preload does not increase the likelihood of fatigue failure because preload, being a constant load, is not a factor in the fatigue load range and does not reduce natural frequency to a level which would be of concern. Total loss of tube preload reduces the tube natural frequency by approximately 15% which is less than the variation in natural frequency within some individual steam generators. Another plant with similar steam generators operates with tube natural frequencies 15% lower than those expected for TMI-1. Thus, the kinetic expansion repair process does nothing to the balance of the tube to increase the likelihood of tube ruptures. (Licensee's Test., fol. Tr. 425, at 6-7; Tr. 482-83; Tr. 499-502).

75. Even if adjacent bowed tubes could come into contact because of compressive loads, such contact could not occur during steady state operation because compressive loads adequate to produce bowing would exist only during heatup, which lasts about 8-10 hours. (Tr. 602-3).

CONCLUSIONS OF LAW

The Board had considered all of the evidence presented by the parties. Based upon a review of the entire record in this proceeding and the foregoing Findings of Fact, the Board concludes that, pursuant to 10 C.F.R. §§ 2.760a and 50.92, the Director of Nuclear Reactor Regulation should be authorized to issue to the Licensee, upon making requisite findings with respect to matters not embraced in this Initial Decision, and subject to the satisfaction of the conditions identified in the Order, infra, an amendment to the operating license which revises the technical specifications to recognize steam generator tube repair techniques, other than plugging, specifically the kinetic expansion tube repair technique.

ORDER

WHEREFORE, IT IS ORDERED, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations, that the Director of Nuclear Reactor Regulation is authorized to issue to the Licensee an amendment to its operating license which revises the technical specifications to recognize steam generator tube repair techniques, other than plugging, specifically the kinetic expansion tube repair technique, upon making requisite findings with respect to matters not embraced in this Initial Decision. Further, this authorization is subject to the satisfaction of Conditions 1, 2 and 6 as set forth in the

Safety Evaluation Report, subject to the satisfaction of Conditions 3 and 4, as modified in Supplement 1 to the SER, and subject also to the satisfaction of the following conditions imposed by the Board in this Order:

1. A duplicate RM-A5 system or suitable equivalent of comparable sensitivity and response time for monitoring radioactive gas in the secondary system shall be installed. The Technical Specifications shall be modified to permit plant operation for a maximum of 28 days with one of these duplicate systems inoperable, and to require plant shutdown if both of these systems are inoperable. As an alternative to the installation of a duplicate system, we direct that the RM-A5 system must be operable at all times during plant operation.

2. In the event of plant operation for an extended period at less than 50% power, the Staff shall require an assessment by the Licensee at the end of 180 days of operation at power levels between 5% and 50%, such assessment to contain recommendations and supporting information as to the necessity of a special eddy-current testing (ECT) shutdown before the end of the refueling cycle. Based on this assessment, the Staff shall determine the time of the next ECT, consistent with the other provisions of the license conditions. In the absence of an assessment, a special ECT shutdown shall take place before an additional 30 days of operation at power above 5%.

Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this Initial Decision shall become effective immediately. It will constitute the final decision of the Commission forty-five (45) days from the date of issuance, unless an appeal is taken in accordance with 10 C.F.R. § 2.762 or the Commission directs otherwise. (See also 10 C.F.R. §§ 2.764, 2.785 and 2.786).

Any party may take an appeal from this decision by filing a Notice of Appeal within ten (10) days after service of this Initial Decision. Each appellant must file a brief supporting its position on appeal within thirty (30) days after filing its Notice of Appeal, (forty (40) days if the Staff is the appellant). Within thirty (30) days after the period has expired for the filing and service of the briefs of all appellants, (forty (40) days in the case of the Staff), a party who is not an appellant may file a brief in support of or in opposition to the appeal of any other party. A responding party shall file a single, responsive brief <u>only</u> regardless of the number of appellants' briefs filed. (<u>See</u> 10 C.F.R. § 2.762).

THE ATOMIC SAFETY AND
LICENSING BOARD

David L. Hetrick

David L. Hetrick
ADMINISTRATIVE JUDGE

James C. Lamb

James C. Lamb, III
ADMINISTRATIVE JUDGE

Sheldon J. Wolfe

Sheldon J. Wolfe, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 31st day of October, 1984.