MAR 27 1992

MEMORANDUM FOR: E. Beckjord, Director
Office of Nuclear Regulatory Research

FROM: J. Hopenfeld
Reactor & Plant Safety Issues Branch
Division of Safety Issue Resolution, RES

SUBJECT: A NEW GENERIC ISSUE: MULTIPLE STEAM GENERATOR LEAKAGE

The enclosed analysis "Safety Issue Relating to Continuous Operation With Degraded Steam Generator Tubes" indicates a core melt probability frequency of 10^{-4}/Ry with containment bypass.

The analysis is submitted for your evaluation and action as appropriate in accordance with RES office letter #1.

/S/

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Enclosure:
As stated

DIST:
RPSIB r/f
DSIR c/f
Hopenfeld:CY
King:CY
Minners:CY
Kniel:CY

RPSIB
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3/26/92

SAFETY ISSUE RELATING TO CONTINUOUS OPERATION WITH DEGRADED STEAM GENERATORS IN PWR's

INTRODUCTION

Recent operating experience in PWR plants, Reference 1, indicates that tight narrow cracks in steam generator tubes are not all being detected with Eddy Current (EC) probes. While such cracks may not leak under normal operating conditions they could leak when subjected to sudden high stresses. The pressure differential across the tube wall which would follow a steam line break accident, SLB, might provide such stresses.

Based on burst tests of sample tubes which were pulled out of service, laboratory data, and analysis, the NRC believes that plant safety is not compromised by the degraded tubes, (Reference 2, 3). The Trojan Plant is allowed to operate with more than 600 hundred defective tubes. The NRC adopted this position on the premise that defective tubes will leak prior to rupture and the leaks will be detected in a timely manner.

This writer believes that the above information, however, is not a sufficient safety basis for continued operation with defective tubes. This concern was documented in a DPO, Reference 4 which in response, the Office of RES requested, Reference 5, that additional information be submitted in accordance with RES Office Letter No. 1.

This document is a response to the above request. Its main purpose is to conduct a preliminary evaluation to show that continuous operation with degraded tubes constitutes a safety risk. This risk, however, can be mitigated by insuring that a sufficient reserve of borated water is available for ECC injection at each plant cite.

ANALYSIS

The fact that degraded tubes neither leak, at normal pressures, nor burst under SLB pressures is not an indication that they will not leak following a SLB accident. The attached RELAP sample calculations (ATTACHMENT 1) show that the total leakage, not leak origin, is the determining factor whether the plant can be brought to a safe shut down. It makes no difference whether the leak origin was from one ruptured tube or many pin hole leaks.

In the example calculations, when the leakage is above 650 gpm, Table 1 and Figures 1-5 show that the leak flow must be terminated in less than eight hours to prevent depletion of the refueling storage tank. Because secondary pressure is near atmospheric following an unisolated SLB, it is difficult to reduce primary pressure below that of the secondary side in a timely manner. Consequently, the leak flow could continue for extended period of time causing the eventual depletion of the RWST. Operator action to minimize the primary pressure can delay the time to exhaust the RWST, however unless the break flow is terminated, a means for replenishing the RWST appears to be the only viable solution.

The frequency of SLB accident outside containment without the ability to isolate the affected steam generator is postulated (Reference 6) to occur at a 10-4 /RY. If a high leakage was to follow it could lead to a core melt because the RWST will be depleted in a period of five to eight hours, as shown in the attachment and discussed above. The NRC also assumes (Reference 6) with a 99.99 certainty that the operator will be able to depressurize the secondary in the above time. The RELAP results, with no credit for operator action, are used in this report. What the operators could do to mitigate the accident will depend on the leak flow rate and is beyond the scope of this study.

The determination of core melt frequency can be obtained by multiplying the probability of leakage following a SLB by the probability of 10-4. The determination of the leak flow from all the degraded tubes requires knowledge of the leakage from each degraded tube and the total number of affected tubes.

From Laboratory data of precrack specimens (Reference 7) one can only conclude that leakage under SLB loads is higher than under normal operating conditions. The cracks in the above specimen were generated in a non prototypical environment and the leak tests were of short duration, therefore, the data cannot be used for leakage estimates in an actual plant, (see Attachment II for additional discussion).

Plant data is not available on leakage of tubes with through the wall cracks at SLB pressures. However, the available plant data suggests that there is a high probability that a leakage will occur but the data is too meager to allow meaningful leak flow estimates. Twenty one specimen which were removed from the Trojan plant, Reference 8, show that the depth of penetration will determine whether the a tube will leak when subjected to high pressure differentials. With the exception of two specimen all the other failed without leakage, on ascent to burst pressures. The two specimen that leaked prior to rupture, however, also exhibited very deep cracks (98% max.). The leak occurred at high than SLB pressure but below the burst pressure. The only conclusion that can be drawn from these twenty one tests is that crack morphology will determine whether a tube will or will not leak at certain applied pressure.

Tube R4-C73 and Tube R21-C22 were pulled from other U.S. plants (Reference 10). Under steady state (delta P) the above tubes leaked between 0 - .3 ml/hr and 0 - << 7 ml/hr. In contrast, under SLB delta P the tubes leaked at a rate of 174 ml/hr and 108 ml/hr. In another case, a tube at a Belgian reactor (R19 - C35) was leak tested in a laboratory and found to leak at a rate of .07 gpm at normal operating pressure and .53 gpm at SLB pressure. The above plant data indicates a high probability of leakage with through the wall cracks and a significant increase in leakage when SLB loads are applied instead of normal operating loads. One may conclude that the probability is one that some leakage will follow a SLB if the defects have penetrated the tube walls.

EC analysis of degraded tubes is more an art than a science and, therefore, a proper evaluation of probe signals require a good knowledge of stress corrosion. In spite of considerable research in this area for the last thirty years the ability to predict crack propagation in the field is still very limited. No practical methods are available to predict probability of leakage from periodic tube inspection. Also, current practice is to shut down the plant when leakage occurs rather than conduct inspection on the predication of a leakage probability. Current RES aging research does not appear to be designed to provide practical information to reactor operators in this regard.

Data on leakage as a function of crack morphology will be required to determine how degraded tubes will behave during the accident. Since experience shows that crack morphology varies not only with location within the tube bundle but also from one reactor to another and from one operating cycle to another the generation of such data is not a practical solution for leakage probability determination. In conclusion there is no way of predicting how many tubes will develop deep micro cracks, how many of them will leak and how much will they leak during an accident.

Attachment 2 provides additional examples of why present data is not usable for leak flow estimates following SLB. The main conclusion drawn from these examples is that the laboratory data used by Westinghouse does not support Westinghouse conclusion that the leak flow following SLB is very small as long as the bobbing coil probe voltage is below 2 volts.

Even if one, for an instant, ignores the question of prototypicality and accepts Westinghouse contention that theirs were valid tests, statistical analysis of the Westinghouse data, Attachment 4, shows that the leak flow rate at the 95% confidence limit and 0 voltage could be .07 gpm. Using the Westinghouse estimate that 680 defective tubes will remain in service, the total leakage per steam generator (50 gpm) is significantly larger than the 0.16 gpm leak rate estimate by Westinghouse. It is not the purpose here to question Westinghouse analysis, but rather to point out that leak rate calculations are very sensitive to model assumptions.

Based on the above discussion a core melt frequency of 10-4/RY may be the best that can be estimated. Plant operators, therefore, must provide assurances that sufficient water to prevent core melt is available to them to avoid RWST depletion. Practically speaking, if a supply of water is available for several days there will be sufficient time to define a solution. Five to eight hours, on the other hand, may not be sufficient.

The first step towards the resolution of this issue is to document the amount of borated water reserves presently available at each plant. It is estimated that one week (NRC+ PLANT) time would be required for this activity for each plant. The corresponding total cost is estimated at $160K ($100K man-yr * 80 plants/50 weeks).

Multiplication of the 10-4 /RY by a dose of 2.7 10+6 (PWR-4 seq) gives 2.7 10+2 man-rem/ry. Assuming 30 yrs remaining life and 80 reactors, we get 50*10+4 man-rem.

CONCLUSIONS:

The present analyses shows that continuous operation with degraded tubes could lead to a core melt due to simultaneous leakage from many tubes following an unisolated steam line break. The risk for such an event can not reliably be estimated because of lack of data. Although a design basis multiple tube rupture could bound the above leakage it is not practical at this time to request the industry to modify present plant designs. The available data does not support NRC position that operation with degraded tube is safe. That position is based on "leak before break" consideration which is acceptable for normal operation but is not applicable to the SLB accident. Public safety will be served by requiring plants to have sufficient borated water reserves on hand. The first step towards this solution is to document present water availability at each plant.

REFERENCES

1. Subcommittee one Material and Metallurgy ACRST-1872, November 6, 1991.

2. Nucleonics week, February 13, 1992.

3. Personal Communication Emmet Murphy (NRR), December 22, 1991.

4. J. Hopenfeld, DPO December 23, 1991.

5. T. Speis to Hopenfeld, memo "Your DPO Dated December 23, 1991", February 19, 1992.

6. NRC Integrated Program for Unresolved Safety Issues A-3, A-4, and A-5, Regarding Steam Generator Tube Integrity, NUREG-0844, April 1985.

7. Steam Generator Tube Integrity Program NUREG/CR-2336; also, WCAP 1313.

8. T. P. Magee et al, Trojan Steam Generator Tubing Destructive Examination Interim Report TR-MCC-186.

9. Steam Generator Tube Integrity Safety Analysis Report for Tube Support Plate Intersections, PGE, December 1991.

10. "Trojan Nuclear Plant Steam Generator Tube Repair Criteria," December 1991 WCAP 13130.

REFERENCES (Con't.)

11. R. N. Parking, Sress Corrosion Crack Coalescence, Corrosion, June 1990, pp. 485.

12. G. Ishack "Steam Generator Tube Degradation - Is it a Safety Concern: Nuclear Eng. International, Jan. 1992.

EVALUATION OF THE TROJAN PLANT ANALYSIS OF LEAKAGE FOLLOWING SLB

In Reference 9 PGE requested the NRC to review their justification for restart of the Trojan plant. PGE concluded that the recently discovered through-wall steam generator tube degradations does not involve unreviewed safety questions (USQ) as described in 10 CFR 50.59 (a) (2). PGE reached these conclusions in reliance, on a Westinghouse study (Reference 10) which examined the consequence of operating steam generators with defective tubes.

Westinghouse concluded that primary to secondary leakage under Steam Line Break (SLB) condition will not constitute a safety problem.

The following analysis shows that primary to secondary leakage during a SLB accident is very sensitive to model assumptions and data source. Different assumptions and different laboratory data leads to drastically different conclusions than those reached by Westinghouse.

2. Westinghouse Analysis, (Reference 10).

The Westinghouse work consists essentially of three parts: 1. Experimental and analytical study of primary to secondary leak rates through cracked tubes under SLB conditions. 2. Determination of crack growth rates and the resultant increase in tube degradation for the next operating cycle.

3. Prediction that a SLB accident at any time during cycle 14 will not exceed 0.12 gpm.

Implicit in the analysis are the following assumptions:

ASSUMPTIONS:

1. SLB leak rates with bobbin voltage indications in the 6 to 50 volts range can be extrapolated to SLB leak rates in the 0 to 1.4 volts range. Figure 8-3, page 8-13, of Reference 10.

2. The laboratory leak rates of Figure 8 are prototypic of SLB leak rates.

3. Successive average voltage changes between three operating cycles determines crack growth rates.

DISCUSSION:

ASSUMPTION 1

The bobbin coil probe voltage depends not only on the total volume of SCC voids in the wall but also on void geometry. Voltage amplitude is not a unique, linear indicator of the propensity of the tube to leak under SLB loads as assumed by Westinghouse. Several, partially through the wall IGA cracks, may give a larger voltage signal than a single deep crack yet the deeper crack, will more likely leak during an SLB accident. The data from Plant A-2 (page 10-11 of Reference 10) indicates that the depth of penetration and not the voltage amplitude determines whether degraded tubes will or will not leak. Tube R4-C73 with, a through the wall crack and a 2.8 volt indication leaked under SLB conditions while another tube with a 2.31 indication but shallower cracks did not leak. (Differences in above voltages are not considered significant within the 40% NDE uncertainty). An indirect indication that crack geometry, not merely crack void affects leakage is provided by the dependence of the burst pressure (Reference 7) on both the depth and the length of the crack.

PGE (Reference 9) stated that the Bobbin probe cannot reliably characterize the depth of penetration of micro flaws. Statistical analysis of laboratory data (Attachment 4 shows that even if the coil probe voltage is zero a finite leakage is possible.

In conclusion, leakage under SLB loads from samples with large voltage signals, is not sufficient to show that through the wall, undetected cracks with indication < 2 volts will not leak.

ASSUMPTION 2

The main variables affecting stress corrosion cracking, SCC, are material condition, temperature, exposure time, environment chemistry and local stresses. While the first three parameters can be simulated in a laboratory it is not practical to simulate water chemistry and local stresses as claimed by Westinghouse.

Experimental data, Reference 11, demonstrate that under cyclic loadings, even small fluctuation in the applied stress rapidly increases the coalesscene of cracks. The Westinghouse specimens were not exposed to the same type of stresses that exist in an operating unit. The interface between the subcooled and saturated region and the foam region are sources for thermal stress fluctuations. Support plates and regions near the tubesheet are subject to local stress fluctuations due to flow induced vibration. The recent SGTR accident at Mihma, is an example (where anti vibration bars which were not installed in the proper locations) of wear tube movement at a support plate.

Chemistry excursions from condenser leaks and primary coolant leaks which occur in operating steam generators are other examples of parameters which cannot be properly duplicated in a laboratory.

Besides the lack of proper environmental simulation, the leakage tests were terminated by Westinghouse after 30 minutes. Considerable industrial experience indicate that high velocity two phase flow can cause material erosion by droplet impingement. The time scale for a jet, emerging into an empty space, from a leaking tube to penetrate a 0.040 inch wall of an adjacent tube is on the order of several hours. (Attachment 3)

The RELAP calculations (Attachment I) indicate that SLB accident with primary to secondary leakage will not be terminated within half an hour as assumed by Westinghouse especially if the leak rate is large. If the accident proceed for several hours, leakage may be increased due to tube to tube damage propagation.

ASSUMPTION 3

As already mentioned above the Eddy Current probe can not characterize the depth of micro cracks in the tube wall. Yet it is the growth and coalescence of these micro cracks that will result in leakage. The average growth rate as obtained by Westinghouse may be related in some manner to these micro cracks but without knowing what that relation is the average growth rate of 45% is simply a guess.

By a way of comment, even if growth rate data was available, the use of average value for leak before break predictions is highly questionable. The proper procedure would be: first establish that cracks grow in a random manner, second obtain a distribution function to the largest crack in a randomly selected tube samples, fourth, use this distribution to predict the time for the largest crack to penetrate the wall.

The Westinghouse assumption that cracks grow at constant rates is in disagreement with the conclusion reached at a recent AEA conference (Reference 12). The applicable conclusions from that conference were that: (1) Inconel 600 is not a stable alloy when used in a steam generator tube material. (2) crack growth rate in tube roll transitions is not constant and is dependent on water chemistry and the particular point in time of the units' cycle.

In conclusion, leakage will be determined by the fastest growing crack in a given tube sample and <u>not by</u> the average crack growth rate as assumed by Westinghouse.

DISCUSSION

Westinghouse efforts in support of PGE request for approval of Trojan restart is focused on showing that primary to secondary leakage following SLB will be within allowable limits. The accomplishment of this task require data which is not available.

Of course the ability to predict leakage per crack is only part of the problem, the other part is the predictions of how many of the cracked tubes will leak when subject to SLB loads. Westinghouse calculates a total of 680 defects which could be left in service. If we accept this number and apply the CE leak data (Ref. 7, pg. 27) to a 0.5 inch long crack, we obtain a total leakage under SLB loads of 6800 gpm instead the .12 gpm per steam generator calculated by Westinghouse. It is obvious that the consequence of SLB is very sensitive to the leakage data employed.