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July 18, 1983
5211-83-207

Office of Nuclear Reactor Regulation
Attn: Harold A. Denton, Director
U. S. Nuclear of Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

Our letter #5211-83-149 of May 9, 1983, submitted Technical Specification Change Request No. 125, which would permit the operation of TMI-1 following repair of the steam generators by any method other than plugging, provided that the repair methods are shown to be acceptable by analysis and approved by the NRC. The letter also requested approval under the provisions of the changed Technical Specification of the precritical non-nuclear heat-up of the plant for testing and the subsequent plant operation using repaired steam generators. The method of repair and supporting documentation were submitted to you in the past.

In his letter of July 13, 1983, H. Dieckamp suggested that the NRC consider the approval of the steam generator repair in two stages: the first being approval of the use of repaired steam generators for precritical non-nuclear hot functional testing, and the second being the ultimate full approval. Such an approval process would permit GPUN to proceed with its test program in a timely fashion, while at the same time, permitting NRC to defer a decision on return to service of the OTSG's until the results of steam generator hot testing have been reviewed.

The original Evaluation of Significant Hazards Considerations submitted with the May 9, 1983, request for approval and its supporting documents is valid for both precritical non-nuclear hot functional testing and subsequent operation. GPUN considers the conclusions drawn to be applicable to both steps of the staged licensing process. Attachment A summarizes the information previously submitted as it relates to the no significant hazards determination.

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In addition, in our meeting with your Staff on July 13, 1983, we were asked for information on current plant conditions that could be used to evaluate the potential consequences of an accident occurring during the precritical test period. Attachment B summarizes these further considerations applicable to the precritical hot functional test period, and are in addition to the items in Attachment A. The low core decay heat (less than 150 kw) after four years shutdown and refueling, the lack of significant amounts of volatile fission products, and the absence of need for the steam generators to provide core cooling render the hazards for precritical operation orders of magnitude less than the FSAR approved analysis.

We hope that the additional information provided will be useful in your review.

Sincerely,

P.R. Clark

P. R. Clark
Executive Vice President

al

cc: H. Silver
J. Stolz
J. VanVliet

ATTACHMENT A

Evaluation of Significant Hazards Considerations, Plant Operational

In order to conclude that a proposed change does not involve a significant hazards consideration, it is necessary to demonstrate the following:

A. There is no increase in the probability or consequences of an accident previously evaluated.

B. The change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The change does not involve a reduction in a margin of safety.

Each of these three items is addressed individually below.

A. The following support the conclusion that there is no increase in the probability or consequences of an accident previously evaluated.

1. We have sufficient understanding of the failure mechanism to assure safe operation and to prevent its recurrence.

- The failure scenario postulated in the failure anlayses is consistent with all the observed features of the cracking phenomenon, the timing of the cracking, the results of metallurgical examinations of tubing taken from the OTSGs and of corrosion tests performed in the laboratory.

- Corrosion tests show that the presence of an active sulfur species is necessary for crack initiation or propagation. Long term corrosion testing with OTSG tubing leading actual plant operation simulating typical operating conditions shows no growth or initiation of cracks.

- Eddy current testing shows that cracks have not been continuing to initiate or propagate.

- The sodium thiosulfate tank has been physically removed from the system.

- Sulfur is now monitored and conservative limits have been placed on its concentration.

- Strict controls have been implemented on potential sources of chemical contamination.

- As much sulfur as possible is being removed from the RCS by chemical cleaning.

- The presence of lithium has been shown in the laboratory to inhibit crack initiation and propagation; minimum lithium concentration limits have been implemented.

Thus, we have concluded that there is not an increased probability of an accident occurring due to new damage.

2. Our inspection techniques have been adequate to find and characterize relevant damage in the steam generators, the remainder of the RCS, and in supporting systems.

- Representative samples of the materials and environments in the RCS and supporting systems were inspected.

- Additional inspections were performed for materials and environments of demonstrated or suspected increased susceptibility.

- Any damaged components identified were repaired or replaced.

- All of the steam generator tubing remaining in service was examined using eddy current techniques.

- The threshold of detectability of the eddy current probes has been identified using calibration standards and laboratory-grown cracks, and confirmed by metallography.

- The threshold of ECT detectability has been demonstrated adequate to identify defects which analysis shows theoretically could propagate to failure due to normal vibration or to transient or accident loads.

- Cracks of a size that would propagate have also been demonstrated by analysis to be detectable due to leakage at operating temperatures and pressures.

- Increased leakage monitoring is planned and new administrative limits on leakage have been imposed to require investigation of leakage which could be from cracks.

- All identified cracks which could propagate or are greater than 40% through wall have been removed from service by plugging the tube or by the kinetic expansion repair techniques.

Thus, we have concluded that the probability of an accident occuring due to undetected damage has not been increased.

3. The kinetic expansion repair technique is adequate to remove from service all significant defects at 8" or higher above the lower face of the upper tube sheet. The repair creates a new tube-to-tubesheet joint below this point which meets the licensing bases of the original joint and which effectively removes degraded portions of tubing at 8" or above.

- In a qualification test program using tubes and tubesheet mock-ups, the kinetic expansion joint was demonstrated

adequate to carry the maximum design basis axial load.

- The joint was demonstrated in the qualification program using mock-ups to continue to meet the licensing bases after simulation of 5 years of thermal and pressure cycling. Longer term testing is continuing.

- The joint was demonstrated in the qualification program using mockups and in full-scale testing in a steam generator at B&W to have minimal and acceptable effect on tube preload, the tube-to-tubesheet weld, the tubesheet, and the adjacent tubes.

- The joint was demonstrated in the qualification program using mockups to create a sufficiently leak-tight seal such that aggregate leakage will not exceed a small fraction of the technical specification limit on leakage.

- The residual stress in the tubing at the transition to the new joint has been demonstrated by laboratory testing and measurement and by analyses to be less than that of the original joint transition.

- The expansion process has been shown in the laboratory testing of tube samples from TMI-1 and of tube sections with laboratory-grown cracks to cause no growth of cracks.

- The expansion process has been shown in full scale testing to have no adverse physical effects on the remainder of the steam generator or other components.

- The expansion process has been shown by analysis and testing to have no adverse chemical effects.

Thus, we have concluded that there is no increased probability of an accident occurring due to failure of the tube-to-tubesheet joint, and further that the act of creating the new tube-to-tubesheet joint has not increased the probability of any other component failure.

4. Use of the repaired steam generator will not change any of the assumptions used to evaluate the consequences of any analyzed accident.

- The qualification program for the new joint has shown, using mock-ups, that the aggregate leakage from all the joints will be a small fraction of that assumed in accident analyses.

- The new administrative limits on leakage will prevent operation with leakage greater than a fraction of that assumed in accident analyses.

- The number and distribution of tubes plugged has been analysed and found to be such that the performance of the steam generator as a heat sink remains within the licensing

basis, during normal, transient and accident conditions.

Thus, the consequences of any analyzed accident are considered not to have been increased by the use of the repaired steam generator.

B. Having concluded that the change will involve no increase in the probability or consequences of an accident, we next consider whether the change creates the possibility of a new or different kind of accident. The following support the conclusion that it has not.

1. The kinetic expansion repair creates a new tube-to-tubesheet joint which meets the licensing bases of the original joint and removes the degraded portion of the tubing from the primary pressure boundary.

- The qualification and test programs described above in item A.3 support the conclusion that the new tube-to-tubesheet joint had no greater probability of failure than the old.

- The new tube-to-tubesheet joint replaces a previous tube-to-tubesheet joint, and does not introduce any new component which could fail in a previously unconsidered manner.

- The potential for future degradation of plugged tubes has been evaluated, and high risk tubes stabilized as well as plugged to prevent interaction of a damaged tube with tubes still in service.

Thus, it is concluded that neither a repaired tube nor a plugged tube introduces either a new type of failure, or a component which is more likely to fail as a result of another transient or accident.

2. As concluded in item A.1, we have sufficient understanding of the failure mechanism to assure safe operation and prevent its recurrence, and we have concluded that the probability of an accident occurring due to new damage is not increased. Thus, it is further concluded that we have not created the potential for any new accidents involving multiple failures of components due to continuing damage.

3. As concluded in item A.2, our inspection techniques have been adequate to find and characterize any relevant damage in the steam generators, in the remainder of the RCS and in the supporting systems. The probability of an accident occurring due to undetected damage is found to have not increased. Thus, the possibility of an unanalyzed accident caused by undetected damage to multiple components is concluded not to have been created.

C. The following supports the conclusion that the change does not cause a reduction in any margins of safety.

1. As discussed in item A.3 above, the new joint meets the licensing bases of the old joint. Thus, margins of safety inherent in the licensing of the joint as part of the primary pressure boundary are maintained.

2. As discussed in item A.2 above, the tubing remaining in service has been confirmed to be acceptable for use under normal vibration and loading, and transient and accident loading. Thus, margins of safety inherent in the licensing of the tubing as part of the primary pressure boundary are maintained.

3. As discussed in item A.2 above, all tubing and other components with significant identified defects have been repaired or removed from service. Thus, no safety margins have been reduced by the use of less effective equipment.

4. As discussed in item A.4 above, operation with the repaired steam generators does not affect any of the assumptions used in determining acceptability of the plant in the licensing bases. Thus, the margins of safety inherent in the FSAR accident and transient analyses are not reduced.

5. In addition, the following steps have been taken to effectively increase the margins of safety:

- Strict administrative limits have been placed on primary to secondary leak rate and changes in leak rate over time.

- Improved chemical controls have been implemented.

- Tube rupture procedures have been reviewed and improved.

- Procedural limits have been placed on steam generator tube-to- shell differential temperature to limit stress on the tubing.

We believe that the information summarized above leads to the conclusion that operation of the plant with repaired steam generators does not:

1. Increase the probability or consequences of any analyzed accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

3. Reduce any margin of safety.

Thus, GPUN concludes that the change does not involve a significant hazards consideration.

ATTACHMENT B

Evaluation of Significant Hazards Considerations
Precritical Non Nuclear Hot Functional Testing

Due to the extended outage, the consequences of any analyzed accident which might occur during the precritical hot functional test period would be considerably less than those calculated in licensing basis documents.

- Decay heat being produced is currently less than 150 kw. The decay heat assumed in design basis analyses is over 1000 times greater than this at the time of scram. Ambient heat loss from the vessel under operating conditions is on the order of 500 kw which is greater than the current decay heat production. Thus a complete loss of heat sink for an extended period of time would not result in core damage. In the event of loss of the steam generator cooling function in its totality, the low levels of decay heat would allow the core to be cooled for extended periods of time with simply the core vessel water inventory and normal thermal losses to ambient. In addition, the decay heat systems are available. While not analyzed in detail, it is also expected that air cooling of the core would maintain the integrity of the fuel element cladding.

- In addition, transients and accidents would be much less severe in scope since decay heat would not be a factor in maintaining RCS temperature and pressure.

- The volatile fission product inventory in the fuel element gap is also extremely small with Xenon-133 and Iodine-131 decayed to almost zero. The total current core inventory of Iodine-129 is less than 2 curies or more than 5 orders of magnitude below the normal core Iodine inventory for FSAR analysis.

- The fission product inventory in the RCS is also considerably less than that considered in evaluating the consequences of an analyzed accident. The failed fuel percentage is a small fraction of the 1% used in FSAR calculations. In addition, many of the isotopes (e.g., I-131, Xe-133) contributing to FSAR analyzed doses have decayed to insignificant concentrations during the extended outage.

  The following isotopes represent the only significant isotopes remaining in the primary coolant, as measured:

Isotope	Concentration (uCi/cc)
Co-60	8.88E-5
Cs-134	1.84E-4
Cs-137	7.90E-4
Sb-125	1.59E-5

  For any transient or accident involving a release to the environment, the extremely low concentration of iodine would essentially preclude a thyroid dose, and whole body doses would be greatly diminished. A direct release to the environment of 500 gpm from the primary system as steam, assuming adverse meteorological conditions, would result in a projected whole body dose of less than 1×10^{-5} mR/hr at the site boundary.