UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 11 1984

MEMORANDUM FOR: Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

FROM: William V. Johnston, Assistant Director
Materials, Chemical & Environmental Technology
Division of Engineering

SUBJECT: EVALUATION OF RECENT INCREASE IN OTSG PRIMARY TO SECONDARY LEAKAGE RATE AT TMI-1

In NUREG-1019 and Supplement No. 1 to NUREG-1019, we found the licensee's OTSG's acceptable for continued operation. By letters dated June 27, July 3, and July 6, 1984, the licensee provided additional information. This information does not affect the evaluation, conclusions or proposed license conditions in NUREG-1019 or Supplement No. 1 to NUREG-1019.

Enclosed is our evaluation.

William V Johnston

William V. Johnston, Assistant Director
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Enclosure: As stated

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Safety Evaluation by
The Office of Nuclear Reactor Regulation
Related to Operation of
Three Mile Island Unit No. 1
General Public Utilities Nuclear Corporation
Docket No. 50-289

Background

In NUREG-1019 and Supplement No. 1 to NUREG-1019, we determined the once-through-steam-generator (OTSG's) had been repaired to their original licensing basis, that the applicable GDCs were met and, therefore, the OTSG's were acceptable for continued operation. By letters dated June 27 and July 3, 1984, the licensee provided additional information on primary-to-secondary leakage rate. By letter dated July 6, 1984, the licensee provided updated information on the long term corrosion test program.

Evaluation

While in a cold shutdown condition, the licensee detected an increase in the background primary-to-secondary leakage rate of the B OTSG. During the September 1983 hot functional test, the background primary-to-secondary leakage rate had been established at 1.0 gph as discussed in Supplement No. 1 to NUREG-1019 at pg. 22. Although the increase in leakage was less than that which would require inspection, the licensee opened the primary side of both OTSG's and conducted bubble, drip and eddy-current test (ECT) examinations. ECT examinations of approximately 150 tubes demonstrated that corrosion was not progressing or initiating. A total of approximately 15 tubes and/or plugs showed indications of minor leakage. Minor leakage of this magnitude is typical from plugs and anticipated from the repaired tubes. The unplugged tubes which were leaking were confirmed to be leaking between the tube and tubesheet, above the kinetic expansion repair joint. It was determined that one tube was leaking past the 6-inch repair joint sufficiently so that background leakage would have increased to an estimated 4 gph at operating conditions. To maintain background leakage low, in addition to the leaking

tube, two additional tubes were plugged for added precaution. Leakage of this type is anticipated as discussed in various sections of NUREG-1019 and Supplement No. 1 to NUREG-1019. The total amount of primary-to-secondary leakage is limited by the plant Technical Specifications and the proposed License Condition No. 4 on pg. 27 of NUREG-1019, Supplement No. 1.

By letter dated July 6, 1984, the licensee updated the long term corrosion test program. Tests which simulate actual plant conditions with water chemistry maintained at maximum anticipated impurity concentrations continue to show no evidence of crack propagation or initiation. The absence of crack propagation or initiation in these tests continue to support the conclusion of NUREG-1019 and Supplement No. 1 to NUREG-1019.

Test Loop No. 1 simulated conditions which assumed that peroxide cleaning would not be conducted and that a continued source of thiosulfate contamination would exist. Under these conditions, one pre-existing crack in a tube specimen taken from the TMI-1 OTSG showed crack propagation. No evidence of corrosion initiation of uncracked tube specimens was detected. These test results verify the applicability of the long term corrosion test program by demonstrating that corrosion continues to propagate in water chemistry conditions where it would be predicted to propagate.

Conclusions

Based on the above evaluation, we find that the information provided by letters dated June 27, July 3 and July 6, 1984 does not affect the evaluation, conclusions, or proposed license conditions in NUREG-1019 or Supplement No. 1 to NUREG-1019.

..e. Mile Island Unit 1 0-289	7/16 SRI Fax/Fire at Information Centar	Three motorcycle riders started a small fire at the fence along the road in front of the TMI Information Center. The fire was quickly extinguished with no significant damage done.
hree Mile Island Unit 1 0-289	7/16 SRI Fax	On 7/16/84, the licensee notified the resident inspector that five (5) steam generator (SG) plugs (3 in SG-A and 2 in SG-B) were missing from the lower (outlet) head area. This plug location verification has been performed following the apparent movement of a plug from tube 148-35 to tube 65-38 in the upper (inlet) head area of SG-A. The licensee's review of the tube plugging records indicate that these plugs were in SG tubes 10-62, 133-77 and 134-73 (SG-A), and 12-51 and 42-16 (SG-B) since April, 1983. Of the 1207 SG tubes plugged in the two SGs, approximately 490 have been plugged using the Westinghouse rolled plugs.
hree Mile Island Unit 1 0-289	7/16 SRI Fax/Seismic Qualification of Diesel Generator Relays	Between 9:30 a.m. and 10:00 a.m., 7/16/84, licensee representatives determined that differential current relays for the Emergency Diesel Generators (EDG's) were not seismically qualified (Category I). The relays lockout the EDG's (output breaker trip and diesel mechanical trip) on a current imbalance in the EDG windings. The relays are manufactured by General Electric for use in Westinghcuse switchgear. Apparently, licensee representatives discussions with these vendors confirmed that the relays installed at TMI-1 were not designed for Seismic Category I use. The licensee made this determination subsequent to the issuance earlier this year of an INPO (Institute of Nuclear Plant Operations) Sigrificant Event Report for a similar problem identified at Palisades. A 10 CFR 50.72 was made at 12:43 p.m., 7/16/84.
ermont Yankee N 50-271	7/16 SRI fax	The licensee disclosed on 7/16/84 that RHR system weld #32-4 was found to have an axial flaw in excess of 10% of the through wall dimensions. RHR line #32 is part of the 20 inch diameter suction line to the RHR pumps for shutdown cooling operations. Weld #32-4 is located inside the drywell. The licensee is still evaluating the type of repair that will be used to fix the flaw necessary. Two additional RHR system welds were added to the sample of welds selected for examination.
ndian Point Unit 2 N 50-247	7/16 RI fax 7/13 ENS call	On 7/23, at 2:45 p.m., with the reactor defueled, inadvertent actuation of train A ESF occurred due to a spurious electrical signal to the safety injection manual actuation relay. Safeguards equipment loads did not sequence onto the 480 Volt busses. Operators manually reset the necessary equipment. The licensee is investigating the cause of the spurious signal.

.icensee/Facility	Notification/Subject	Description of Items or Events
.imerick 'IS 50-352/353	Information Item	Mr. Chilk, Mr. Bates, and Mr. Mazuzan, will go on an information tour of the Limerick site with the Senior Resident inspector on Tuesday, July 17, 1984.

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MORNING REPORT - REGION V
DATE: JULY 17, 1984

FACILITY	NOTIFICATION	ITEM OR EVENT	REGIONAL ACTION
SOUTHERN CALIFORNIA EDISON COMPANY SONGS 3 DN 50-362	TELEPHONE CALL FROM RESIDENT INSPECTOR ON 7/16/84	STEAM GENERATOR TUBE LEAK: A TUBE LEAK IN STEAM GENERATOR E-089 WAS MEASURED AT 33 GALLONS PER DAY ON JULY 16, 1984 AND AT 68 GALLONS PER DAY (0.05 GALLONS PER MINUTE) ON JULY 17,1984. THE REACTOR COOLANT GROSS ACTIVITY IS 14 UCI/GRAM WITH A DOSE EQUIVALANT IODINE ACTIVITY OF 0.35 UCI/GRAM. THE TECHNICAL SPECIFICATION LIMIT FOR STEAM GENERATOR TUBE LEAKAGE IS 720 GPD FOR ONE S/G OR 1 GPM TOTAL. A 1 GPM TUBE LEAK WOULD RESULT IN A DOSE RATE AT THE SITE BOUNDARY OF ABOUT 0.005 MREM/HR. THE LICENSEE IS CLOSELY MONITORING THE SITUATION AND PLANS TO SHUT THE UNIT DOWN WELL BEFORE THE TECH SPEC LIMIT IS REACHED. THE RESIDENT INSPECTOR IS ALSO ACTIVELY INVOLVED IN FOLLOWING THIS MATTER.	FOLLOWUP PER MC 2515

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