ENCLOSURE

SAFETY EVALUATION REPORT

REVIEW OF SUPPLEMENT 1 TO WCAP-10698,

EVALUATION OF OFFSITE RADIATION DOSES FOR

A STEAM GENERATOR TUBE RUPTURE ACCIDENT

INTRODUCTION

In a May 24, 1985 letter to the NRC, the Steam Generator Tube Rupture (SGTR) Subgroup of the Westinghouse Owners Group (WOG) submitted Supplement 1 to WCAP-10698, Evaluation of Offsite Radiation Doses for an SGTR Accident, to support the resolution of the licensing issues associated with an SGTR accident. This Safety Evaluation Report documents the s' ff review of the results and methodology presented in Supplment 1 to WCAP-10698.

As a result of the January 1982 SGTR at the R. E. Ginna Plant, the NRC has questioned the assumptions used in the safety analysis of a design basis SGTR, including the operator action time assumed in terminating leakage from the primary to the secondary coolant systems, and the qualification of the equipment assumed to be used in the SGTR recovery. In response to these concerns, a subgroup of utilities in the WOG was formed to address the licens' g issues associated with an SGTR event on a generic basis. In December of 1984, the subgroup submitted WCAP-10698, SGTR Analysis Methodology To Determine the Margin to Steam Generator Overfill, which presented the development of a design basis SGTR analysis methodology. Supplement 1 to WCAP-10698 presents the evaluation of potential offsite doses for a design basis SGTR in the absence of steam generator overfill. The subgroup also plans to submit by November of 1985 an evaluation of the consequences of steam generator overfill resulting from an SGTR.

WCAP-10698 presented results from the following tasks in the development of a design basis SGTR analysis methodology: (1) development of LOFTTR1, an analytical model which is a modified version of LOFTRAN, that incorporates improved models for break flow and the steam generator secondary side, and an improved capability to simulate the operator actions for SGTR recovery; (2) determination of operator action times for design basis application based on the guidelines of Revision 1 of the WOG Emergency Response Guidelines issued in September 1983; (3) sensitivity studies to identify conservative values of plant parameters; (4) single failure analysis of the design basis equipment; and (5) application of the methodology to a reference plant.

The evaluation of offsite doses presented in Supplement 1 to WCAP-10698 used steam release rates to the environment and thermal and hydraulic parameters for the primary and secondary sides which were calculated using the LOFTTR1 computer code, and operator action times developed in WCAP-10698. In addition, the single failure analysis and sensitivity studies of Supplement 1 relied heavily upon the corresponding results of WCAP-10698. It should also be noted that staff review of the subgroup's evaluation of the consequences of steam generator overfill could potentially lead to changes in the analysis assumptions used in evaluating the radiological consequences of a design basis SGTR accident. Thus, the results and conclusions of this SER will be modified as appropriate if staff review identifies the need for significant changes in the design basis SGTR analysis methodology presented in WCAP-10698. WCAP-10698 is currently under review by the staff with SER issuance for WCAP-10698 and for the evaluation of overfill consequences projected for the second quarter of FY1986. It should be noted, however, that the review of the dose analysis methodology (Section 5.0) presented in Supplement 1 with its assumptions and models of coolant activity levels and iodine transport processes is not dependent upon the results of the review of these other submittals.

Supplement 1 to WCAP-10698 presents the results of the following tasks: selection of a reference plant and site; single failure analysis to determine the worst single failure with respect to offsite doses; calculation of the mass releases to the environment using the results of the LOFTTR1 analyses from WCAP-10698 for mass releases prior to termination of the primary to secondary leakage, and the results of an analysis based on a continuation of the SGTR recovery actions in the WOG Emergency Response Guidelines for mass releases during the period between leakage termination and the end of the accident; and the development of the dose analysis methodology.

DISCUSSION

The evaluation of offsite doses in Supplement 1 to WCAP-10698, was performed for a reference plant and site. Atmospheric dispersion factors which were representative for typical Westinghouse plants were used in the dose calculations. The reference plant, as described in Section 4.1 of WCAP-10698, was selected on the basis of a preliminary analysis which provided estimates of the relative time to overfill for several representative Westinghouse plant types. The calculations to determine the relative time to overfill compared the secondary side steam volume to the equilibrium break flow rate, defined as the break flow rate at the primary pressure at which outgoing break flow is balanced by incoming safety injection flow. The calculations did not consider accident system response and operator actions.

The staff notes that the selection of a reference plant based on the above estimates of the relative time to overfill does not assure the selection of the most conservative plant design with respect to potential offsite doses. Operator action time and system response time, which depend on plant specific equipment, operating procedures and individual plant design and parameters, must be considered in determining the duration and severity of the accident and the amount of radioactivity released to the atmosphere. The evaluation presented in Supplement 1 to WCAP-10698 was based on a reference plant with representative atmospheric dispersion factors, instead of a conservative plant design with bounding atmospheric dispersion factors. The staff concludes that the offsite dose calculations presented in Supplement 1 constitute representative examples of the application of the proposed design basis SGTR analysis methodology to a reference plant and site, but are not bounding cases. Plant specific analyses will be necessary to demonstrate that the radiological consequences of a postulated SGTR accident at an individual plant meet the acceptance criteria of Section 15.6.3 of the Standard Review Plan (NUREG-0800, Rev. 2, July 1981).

The single failure analysis to determine the worst single failure with respect to offsite doses and sensitivity studies to identify conservative (with respect to offsite doses) plant conditions, parameters, and other analysis assumptions presented in Supplement 1 relied heavily upon the results of the single failure analysis and sensitivity studies in WCAP-10698 which were used to identify conservative assumptions with respect to margin to overfill. (The margin to overfill is defined as the steam space volume remaining below the steam generator outlet nozzle when the primary to secondary leakage is terminated). As stated in Supplement 1, it is expected that most of the conservative assumptions and initial conditions which were used in the evaluation of the margin to overfill would also be conservative with respect to offsite doses. This is based on the fact that both offsite doses and the potential for overfill are primarily dependent upon the amount of primary to secondary leakage and the amount of steam released from the ruptured steam generator.

The staff agrees that, in general, conditions and assumptions which are conservative with respect to overfill would also be conservative for offsite doses. The decrease in the margin to overfill as a result of a postulated single failure or a conservative analysis assumption is due to the increased operator action time and system response time required to complete the recovery action. The increased operator action time and system response time would prolong the accident and generally lead to increases in the release of radioactivity to the environment.

As discussed in Supplement 1, however, a decrease in the margin to overfill represents the additional net accumulation of water in the secondary side of the ruptured steam generator. Net accumulation of water increases with increases in the amount of primary to secondary leakage, but decreases with increases in the amount of steam released from the ruptured steam generator. (This follows from mass continuity considerations if one neglects interdependency effects.) For those cases in which the amount of steam released to the atmosphere does not change, conservative conditions with respect to overfill would also be conservative with respect to offsite doses. In these cases the decrease in the margin to overfill is a result of an increase in the amount of primary to secondary leakage due to increased operator action time and system response time. This prolongs the accident and results in increased releases of radioactivity to the environment.

The single failure analysis presented in Supplement 1 has identified and examined those cases which result in increases in the amount of steam released from the ruptured steam generator. In addition, the analysis identified an estimated proprietary hydraulic parameter which was conservative with respect to offsite doses, but was not conservative with respect to margin to overfill. This assumption is discussed in Section 5.2 of Supplement 1 and was investigated in various case comparisons, including a comparison of calculated doses for Cases 1 and 5.

Based on the above findings, the staff concludes that the single failure analysis and sensitivity studies in Supplement 1 have identified the worst single failure and the analysis assumptions which are conservative with respect to offsite doses. This conclusion is based upon the following: staff review of the sensitivity studies and equipment failure evaluation in WCAP-10698 to assure that conservative plant conditions, parameters, and analysis assumptions and the worst single failure with respect to margin to overfill have been properly identified; the generic applicability of the single failure analysis in WCAP-10698; and the use of the assumption which was identified in Section 5.2 of Supplement 1 to be conservative with respect to offsite doses but not with respect to margin to overfill in subsequent applications of this methodology for the evaluation of offsite doses from an SGTR accident.

The results and conclusions of this SER will be modified as appropriate if the review of WCAP-10698 identifies the need for significant changes in the results of the sensitivity studies and equipment failure evaluation presented in WCAP-10698. In addition, the single failure analysis presented in WCAP-10698 is based on the WOG Emergency Response Guidelines which are applicable to nearly all Westinghouse plants, and a design basis equipment list which identifies sufficient principal equipment to terminate primary to secondary leakage for all Westinghouse plants. The generic applicability of the analysis may be limited, however, based on plant specific differences which would affect changes in operator action times and system response times required to complete the recovery operation as a result of a postulated single failure. For example, the staff notes that the results of the single failure analysis in WCAP-10698 may not apply to two loop Westinghouse plants. If, as a result of the staff review of WCAP-10698, it is determined that the single failure analysis is not generically applicable, then plant specific analysis to determine the worst single failure with respect to offsite doses may be required.

The staff has reviewed the evaluation of offsite doses for the single failure cases considered in Supplement 1 to WCAP-10698. Mass releases from the ruptured and intact steam generators to the atmosphere were determined from LOFTTR1 analyses (described in WCAP-10698) for the period from accident initiation to the termination of primary to secondary leakage. Mass releases for the period from leakage termination to the end of the accident, assumed to be 8 hours, were determined from an analysis based on SGTR recovery operations in the WOG Emergency Response Guidelines. Revision 1 of the Emergency Response Guidelines provides for three alternate means of performing the post - SGTR cooldown. The method using steam dump, Guideline ES-3.3, was selected for evaluation of the mass releases since it results in conservative results for the offsite dose evaluation. The ES-3.3 guideline specifies the actions required to bring the Reactor Coolant System down to Residual Heat Removal System temperature and pressure levels. This is accomplished by using steam dump to the condenser, or using the power operated relief values of the intact and ruptured steam generators if the condenser is unavailable.

The dose analysis methodology as presented in Supplement 1 to WCAP-10698 uses assumptions for the initial primary and secondary coolant activity concentrations, the radiological consequences of iodine spiking, a coolant iodine spiking model for the accident initiated iodine spike case, and primary to secondary-system leakage in the intact steam generators which are consistent with those in Section 15.6.3 of the Standard Review Plan. In the determination of iodine transport to the atmosphere, the methodology presented in Supplement 1 discusses the volatilization of iodine in the primary coolant due to flashing and atomization, and the scrubbing of iodine contained in the steam phase and atomized droplets for release points which are below the steam generator water level. It does not, however, explicitly describe the models and assumptions used in the determination of iodine transport in the faulted generator. Thus, no staff review of the iodine transport models was possible, and independent staff verification using the iodine transport models referenced in the Standard Review Plan will be necessary on a case-by-case basis. It is the staff's position that plant specific analyses should provide a detailed description of, or reference, the explicit iodine transport models used in the analyses.

The staff concludes that the dose analysis methodology presented in Supplement 1 to WCAP-10698 is generally consistent with Section 15.6.3 of the SRP and, thus, is acceptable with the exception of the iodine transport models which will be reviewed on a case-by-case basis.

CONCLUSIONS

The staff has reviewed the methodology and results presented in the evaluation of offsite doses for an SGTR accident in Supplement 1 to WCAP-10698. The staff concludes that the dose analysis methodology used in the evaluation is acceptable with the exception of the determination of iodine transport to the atmosphere for which explicit models and assumptions were not provided. Independent staff verification using the iodine transport models referenced in the SRP will be necessary on a case-by-case basis.

The staff notes that the offsite dose calculations presented in Supplement 1 were based on a reference plant and reference site and, thus, did not constitute bounding cases for all reactors and sites. Plant specific analyses will be necessary to demonstrate that the radiological consequences of a postulated SGTR accident at an individual plant meet the acceptance criteria of Section 15.6.3 of the SRP.

The results and conclusions of this SER will be modified as appropriate if staff review of WCAP-10698 and of the subgroup's evaluation of the consequences of steam generator overfill identifies the need for significant changes in the design basis SGTR analysis methodology presented in WCAP-10698.

IMPLEMENTATION

As discussed above, plant specific evaluations of offsite doses using appropriate plant specific mass releases and thermal and hydraulic parameters for the primary and secondary systems will be necessary for individual plants. The evaluation should consider the worst single failure and plant conditions, parameters, and assumptions which are conservative with respect to offsite doses. The results of the single failure analysis and sensitivity studies in Supplement 1 are acceptable, provided the single failure analysis in WCAP-10698 is generically applicable and the staff review of WCAP-10698 does not identify the need for significant changes. If, as a result of the staff review of WCAP-10698, it is determined that the single failure analysis is not generically applicable, then plant specific single failure analyses to determine the worst single failure with respect to offsite doses may be required. In addition, the plant specific evaluations of offsite doses should use the analysis assumption which was identified in Section 5.2 of Supplement 1 to be conservative with respect to offsite doses but which was not conservative with respect to margin to overfill.

The plant specific analysis should provide sufficient information for staff review, including the following information as a function of time during an SGTR, to allow an independent evaluation to be made by the staff of the radiological consequences:

(1) Total mass releases and mass release rates from the ruptured steam generator to the atmosphere,

(2) Total mass releases and mass release rates from the intact steam generator(s) to the atmosphere,

(3) Primary to secondary system leakage flow rate in the faulted generator (break flow rate),

(4) Pressure differential between the RCS and the ruptured steam generator,

(5) Water level above the break location in the ruptured steam generator,

(6) Mass of water in ruptured steam generator,

(7) Pressure in the ruptured steam generator, and

(8) RCS hot leg and cold leg temperatures in the ruptured loop.

In addition, it is the staff's position that plant specific analyses should include a detailed description of, or reference, the explicit iodine transport models used in the analyses.

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

AUG 1 1 1987

MEMORANDUM FOR: C. Y. Cheng, Acting Chief
Materials Engineering Branch
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THRU: Keith Wichman, Section Leader
Materials Engineering Branch
Division of Engineering and Systems Technology

FROM: Herbert F. Conrad
Materials Engineering Branch
Division of Engineering and Systems Technology

SUBJECT: TRIP REPORT - NORTH ANNA 1 STEAM GENERATOR TUBE RUPTURE INVESTIGATION, JULY 22, 23, 29 AND 30, 1987

Summary

The 360° circumferential double ended break at the top of the uppermost cold leg tube support plate is now believed by the licensee to be related to both stress corrosion cracking and fatigue with the origin (ID or OD) not yet known. The fracture location was last inspected in 1981; no crack indications were found at that time. The Utility has committed to a comprehensive full tube length, all steam generators inspection that will be the most extensive and sensitive eddy current inspection program conducted on a U. S. Nuclear Plant to date. Every effort will be made to remove a sample of the fractured tube, but its location within the bundle at the top near the U-bend (row 9, column 51) makes removal and stabilization of the remaining tube end difficult. The plant was shut down in an orderly manner after the rupture with all safety limits and thermal margins maintained. The licensee's analysis indicates that the event was bounded by the steam generator tube rupture event calculations in the Plant Final Safety Analysis Report. Radioactive releases via the condenser air ejector were less than 1% of the Technical Specification Limit and well within 10 CFR 100 limits.

Introduction

I traveled to the North Anna Nuclear Power Plant on July 22, 1987 and joined with Dr. C. V. Dodd, Oak Ridge National Laboratory, to participate on the Augmented Inspection Team (AIT) which was led by Floyd S. Cantrell of Region II. Dr. Dodd is under a technical assistance contract with the Materials Engineering Branch for on-call consultation in the area of eddy current testing. He also does research for the Office of Nuclear Regulatory Research. We participated in the AIT Team activities on July 22 and 23 and returned to North Anna on July 29 for the meeting between North Anna management and J. Nelson Grace, Region II Administrator and members of NRR management. On July 30 we completed our input to the AIT Inspection Report covering eddy current testing.

D/19

Steam Generator Tube Rupture

On Wednesday, July 15, 1987, at approximately 6:30 a.m., Unit 1 of the North Anna Power Station experienced a tube rupture in steam generator C, of tube R9C51 at the top of the seventh support plate in the cold leg. This accident occurred only about 24 hours after the reactor returned to 100% power after the Spring 1987 refueling outage. The exact chronology of this tube rupture event is given in Attachment 1. The operators at the plant were able to bring the reactor to a cold shutdown mode without further damage to the plant or any significant radiation release to the environment due to appropriate execution of the operating procedures. A review of the event, run on the training simulator by the operators for the NRC staff, demonstrated the shutdown process after the tube rupture. In addition to the training simulator, a model power plant in a see-through glass case was shown. A simulated tube rupture showed the loss of coolant in one steam generator and the problems and effects of this on the plant. The model had all the major components of a nuclear power plant, including two steam generators, one once-through and one recirculating steam generator. The water level and boiling in the various components could be seen.

At the time of the initial meetings with the power station personnel, the exact nature of the defective tube was not known. Dr. Dodd and I were present in the Westinghouse trailer on Tuesday afternoon (July 21) when the eddy current tapes of the leaking tube were analyzed for the first time. They showed an indication at the top of the seventh tube support plate so large that it saturated out the electronics. The analysist insisted that it had the signature of a tube end. The utility at that time, however, reported it as a ½" to 1" long longitudinal crack even though calculations indicated that such a short crack could not account for the observed leak rate (560-637 gpm). It was not until the video fiber-optics examination Tuesday night that the tube was confirmed to be a 360° guillotine break with the ends approximately ½" to 1" apart. Detailed examination of the videotape examination of the fiber optics scan of the tube by the VEPCO Metallurgist is given in attachment. My own observations agree with his.

The full length of the tube was inspected in 1979 and again in 1981 by a bobbin probe. It was inspected during the April 1987 refueling outage, only to the seventh support plate on the hot leg side, not the full length or around to the seventh cold leg support plate. The review of the 1981 inspection tapes revealed nothing. These tapes are analog and the present inspection equipment (MIZ18) can give a far superior inspection. An investigation of the background of the previous eddy-current inspections will be performed.

Background of Eddy-Current Tests

The generators were modified before operation by explosively expanding the tubes in the tubesheet region, eliminating the crevice region that had been a source of tube leaks at other plants in the 1970s. This, however, moved the expansion region up the tube near the top of the tubesheet, which has caused some eddy-current inspection problems. The history of eddy-current inspections and repairs is summarized in Attachment 3. The inspections in 1979 with the bobbin type coils revealed that denting had occurred. This denting has considerably complicated the subsequent eddy-current tests of the tubes at the intersection of the tubes and the tube supports and made the detection and measurement of other modes of degradation much more difficult. Profilometry data performed in subsequent inspection has revealed no growth in the denting, but leaks have revealed the continued degradation of the generator. The early eddy-current inspections were performed with single-frequency equipment and recorded on analog tape.

More accurate inspections, performed with three-frequency instruments using digital data reduction and analysis techniques, revealed what was referred to as "distorted tube support plate indications." These were first observed in 1984, and attempts to resolve these indications led to the use of the 8 x 1 probe and the rotating pancake coil (RPC), described in Attachment 4. Inspections with these probes resolved the distorted tubesheet signals into axial cracks for some of the tubes, with the others found to have no defects. In addition to these, circumferential defects were located in the tubesheet expansion region. These defects were detected by the 8 x 1 probe and verified and mapped by the RPC. About 150 tube support junctions in steam generators A and B were also inspected with the RPC. These intersections had not revealed any indications with the bobbin coil inspection, and they did not reveal any indications in the RPC inspection. Tube pull data from the 1985 and 1987 outages revealed that there was intergranular cracking (IGC) at the top of the first tube support plate, on the outer diameter of the tube, up to 28% deep. In addition, there were circumferential cracks on the tube inner diameter at the top of the tubesheet, associated with the explosive expansion and axial cracks at the tube supports, associated with the dents. Tube burst tests on a pulled tube having an 84% defect, 180° around the tube, showed that 10,700 psi was required to fail the tube.

Eddy-Current Inspection Plan

After the failure of tube R9C51 in a circumferential manner at the top of the seventh tube support in the cold leg, an extensive eddy-current testing program was planned with emphasis on detecting circumferential defects. This program is listed in Attachment 5 and includes the inspection of every tube support junction (and the straight tube sections in between) in all three steam generators with an 8 x 1 pancake array probe. This is the most extensive, sensitive, and ambitious inspection program attempted to date for steam generator inspection. It will strain the availability of probes and data analysts in the industry. This probe (8 X 1) has the sensitivity to detect all inner diameter defects, either axial or circumferential, 20% or deeper, with a length of 3/16 in. or longer. In addition, it should also be able to detect outer diameter cracks and intergranular attack on either the inner or outer diameter. All indications detected by the 8 x 1 probe will also be tested using the RPC probe. The tubing standard used for the pancake coils has a range of outer diameter circumferential electrodischarged machined notches ranging from 20 to 100%. The standard scans showed good depth separation between the outer diameter notches of different depths at 400 kHz, and a good separation between the tube support signals and defect signals at 200 kHz, although some of the depth measurement ability was lost at this lower frequency. Although no notch standard was available for inner diameter defects, they could certainly be detected and estimated from an interpretation between no defect and 100% defect.

Evaluation of Procedures and Analysis

We obtained a copy of the North Anna 1, "Analysis Rules-Steam Generator Inspection Procedure Package" dated July 1987 and Dr. Dodd, ORNL, the NRC's eddy current consultant reviewed the written procedures as well as observing the actual eddy current data analysis in the Westinghouse Trailer at the North Anna Site. He provided the following evaluation:

> "The written data analysis methods are clear and detailed, with more than adequate examples for all three types of eddy current inspections. The senior data analysts are very experienced with the facility, the equipment, and the general types of tube degradation that has occurred at all other Westinghouse facilities and with the methods of detecting tube degradation. The Intelligent Eddy Current Data Analysis System (IEDA) is being used as an aid in flagging suspect bobbin coil indications which are then dispositioned by the data analyst. The data from each tube is independently reviewed by two different analysts, with one using the Westinghouse IEDA system and the other using a Zetec Digital Data Analysis System (DDA4). All the data analysts are at least certified Level II, American Society of Nondestructive Testing (ASNT) in accordance with ASNT requirements. This includes industry experience, class room training, a technical education, and testing on both general eddy current knowledge and specific eddy current knowledge for steam generator inspection. The analysts are given additional training by Westinghouse and are required to pass a test that covers the specific data analysis used for the three eddy current tests at North Anna 1."

Current schedules call for return to power on September 30, finish of inspection on September 5 and for the removal of R9-C51 to begin on August 11 by shrinking the tube with a longitudinal weld bead. The Utility plans to issue daily inspection status reports, the latest of which is included as Attachment 6. I will keep you informed of all new developments.

H.F. Conrad

Herbert F. Conrad
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cc: See next page

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As stated

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ATTACHMENT 1

CHRONOLOGY
STEAM GENERATOR TUBE RUPTURE EVENT
NORTH ANNA POWER STATION UNIT 1
JULY 15, 1987

To support the investigation of the Steam Generator Tube Rupture the following chronology was reconstructed from the print outs of the alarm typewriter attached to the Control Room P-250 process computer, the Sequence of Events Recorder (Dranetz) driven by the Hathaway annunciator system, the data printouts extracted from the record kept by the ERF Computer in the Technical Support Center, RO and SRO logs and interviews, and strip charts from Control Room recorders.

Selected data was transmitted from the various records based on the significance of each datum as it identified a sub-event or demonstrated, explicitly or implicitly, a sub-event in the sequence. The intent is that this chronology can be integrated with other analysis to determine the timeliness, accuracy and effectiveness of the measures applied to mitigate the accident.

Once the data was transcribed, a review was performed to identify the synchronism for time of the various data sources. The principal item selected for synchronism was the Automatic Pzr Lo-Lo SI. The SI action incorporates several actions including feedwater isolation and normal charging isolation that make it readily comparable over all records. The Sequence of Events Recorder logged SI at 06:35:24:805; the alarm typewriter on the P-250 logged SI at 0639. However the earlier Reactor Manual Trip has caused the P-250 to alter its scan rates. The P-250 Post Trip review logged SI at 06:35:24 plus 1012 cycles, which equates to 06:35:40.86. The ERFC data set collected at 06:34: 14 records full normal charging flow and full power feed flow to the steam generators, approximately 16 seconds after the reactor trip had been manually initiated. By 06:34:21, the ERFC da set charging flow is reduced to 82.568 gpm and feed flows are about 600KLBH to 800KLBH. By 06:34:34, all flows had reached a stable but low level. It appears that SI occurred at or slightly before 06:34:14. This chronology will use 06:34:14.

For automatic initiation of Safety Injection, the clock comparisons are as follows:

RECORDER	TIME
Sequence of Events Recorder (SER)	06:35:24:805
P-250 Computer (Alarm Typewriter)	06:35:41
ERF Computer (ERFC)	06:34:14

For Reactor Manual Trip the clock comparisons are as follows:

RECORDER	TIME
Sequence of Events Recorder (SER)	06:35:04:548
P-250 Computer (Alarm Typewriter)	06:35:24
ERF Computer (ERFC)	06:33:56

It is concluded that the P-250 led, the SER was within 20 seconds of the P-250 and the ERFC was about one minute behind the P-250.

The chronology that follows is annotated by clock time based on the P-250. All the events that occurred within each minute are listed in order of occurrence as could best be determined.

CHRONOLOGY
STEAM GENERATOR TUBE RUPTURE EVENT
NORTH ANNA POWER STATION UNIT 1
JULY 15, 1987

July 14, 1987

2238 Air Ejector Radiation Monitor (RM-RMS-121) was declared inoperable due to erratic operation.

July 15, 1987

0630 An alarm was received on the Unit 1 annunciator panel for Main Steam (High Range) Radiation Monitor. When checked by backboard operator, "A" and "B" monitors were in "Alert" and "C" monitor was in "High" alarm.

0631 Unit 1 CRO observed the pressurizer level decreasing rapidly.

0632 U-2 SRO recalled the Shift Supervisor and U-1 SRO to the Control Room. U-1 CRO took manual control of charging and set FCV-1122 to full open. Received Pressurizer low pressure alarm at 2135 psig (Alarm Typewriter).

0633 Shift Supervisor entered the Control Room and directed letdown isolation. CRO initiated realignment of charging pump suction to RWST and a 2% per minute turbine ramp down. Third CRO (Backboards Operator) assumed BOP duties on Unit 1 Control Board to assist Unit 1 CRO. (Alarm Typewriter: Make-up commenced. VCT low level alarm 20.3%.)

0634 STA arrived in the Control Room.
Alarm Typewriter: Pressurizer Pressure 2109 psig.
Superintendent of Operations was notified and directed the Unit to be manually tripped.

0635 At direction of Shift Supervisor, U-1 CRO manually tripped the reactor and turbine and initiated EP-0. CRO observed pressurizer level at approximately 45% and pressurizer pressure at approximately 2100 psig at the time of the manual trip.
POST TRIP REVIEW: Initial Event 06:35:24

Rx Manual trip (2)	0.00 sec.
Turbine Trip and P7	0.16
High Flux Rate Trip	0.33
Rx Manual trip (1)	0.60
Pzr - Lo Press Trip	2.80
Stm Gen B Lo-Lo Trip	4.2
Stm Gen C Lo-Lo Trip	4.5

Stm Gen A Lo-Lo Trip 4.5
Pzr Lo-Lo SI 16.86
Manual SI Train "A" (1) 43.38
Manual SI Train "B" (2) 44.97 seconds

Alarm Typewriter: Auxiliary Feed Water Pumps Start. VCT level 22.1% increasing (indicates that charging pump suction shift to RWST is completed.)
SER: Main Feedwater Pumps Trip (06:35:25)

0636 • Unit 1 CRO noted pressurizer pressure less than 1700 psig and pressurizer level less than 5%.
Alarm Typewriter: Pressurizer level 2.7%. Main Feed pump Breakers tripped. "B" Charging Pump Start. "A" and "B" LHSI pumps start.

0637 Alarm Typewriter: G-12 breaker open.

0639 A Notification of Unusual Event was declared. Unit 2 SRO assumed duties as Interim Station Emergency Manager and initiated the EPIP's. (Step 21 of EP-0)

0640* Entered EP-3 from Step 23 of EP-0.

0641 Alarm Typewriter: T_{AVG} less than 543°F. P-12 interlock set.

0642 Alarm Typewriter: "C" Steam Generator level increasing above 18% narrow range.

0644 Alarm Typewriter: SI and Phase A reset. LHSI pumps "A" and "B" shutdown. (Steps 9, 10, and 13, respectively of EP-3)

0645 Alarm Typewriter: "C" Steam Generator at 25% narrow range and increasing.

0646* Auxiliary Feed Water to "C" Steam Generator isolated. (Shift Supervisor confirmed Steam Generator Tube Rupture in "C" Steam Generator based on "C" Steam Generator level continuing to rise.)

0646 ERFC: "C" Main Steam Trip Valve closed.

0647 Alarm Typewriter: "A" Steam Generator at 23% (Narrow Range) and increasing.

0648* Steam supply from "C" Steam Generator to 1-FW-P-2 (Terry Turbine) isolated. (Step 4 of EP-3)

0648 ERFC: Graphs of pressurizer level and RCS pressure reveal increasing level and pressure. (Also noted on the strip chart in the Control Room.)

0649 Alarm Typewriter: Pressurizer Lo Press/Steam Line High Flow SI circuit blocked. (The Note prior to Step 15 of EP-3) Commenced rapid cooldown on "A" and "B" steam dump valves. (Step 15 of EP-3) Alarm Typewriter: "B" Steam Generator level at 25% (Narrow Range) and increasing. Alarm Typewriter: Pressurizer level 8.5% and decreasing.

0650* Unit 1 CRO noted pressurizer level off scale low.

0651 Initial notifications made to the State/Local Governments (EPIP 2.01) and NRC (EPIP 2.02).

0652 Alarm Typewriter: "A" T_c = 509.5°F "B" T_c = 509.5°F "C" T_c = 523.5°F

0654 Interim Station Emergency Manager upgraded event classification to "ALERT". (Step 40 of EP-3)

Alarm Typewriter: "B" Main Feed Pump breakers racked to test and closed (To provide a flow path for condensate pumps to feed "A" and "B" Steam Generators) Steam Generator "A" and "B" pressures at 589 psig.

0655 Initiated EPIP-3.01, 5.03, and 5.04 (Call Out, Accountability, and Access Control).

0657 Strip Chart: RCS Temperature being maintained at 480°F (Step 15 of EP-3). Alarm Typewriter: Pressurizer Spray control at 100% demand. Valves "A" and "B" open (step 18 of EP-3).

0658* Unit 1 CRO noted pressurizer level on scale and increasing.

0659 Alarm Typewriter: Source Range Nuclear Instruments manually re-energized. (Intermediate Range Nuclear Instruments were undercompensated.)

0700 Alarm Typewriter: Pressurizer low level heater cut off cleared (level at 15% and increasing). Pressurizer heaters energized (835 KW).

0701 Alarm Typewriter: Unit 1 CRO manually de-energizes pressurizer heaters.

0702 Notifications made to the State/Local Governments and NRC of upgraded alert classification.

0704 Alarm Typewriter: Opened one Pressurizer PORV to reduce pressure (Step 19 of EP-3). SRO observed pressure reduction of approximately 40 psig and instructed CRO to close PORV and spray valves (Steps 18 and 19 of EP-3). Alarm Typewriter: Pressurizer Relief Tank pressure 15 psig.

SRO noted "C" Steam Generator level increase stopped.

SI reduction criteria met (Step 21 of EP-3). "B" Charging Pump secured (Step 22 of EP-3).

Initiated the isolation of BIT flowpath (Step 24 of EP-3) and established the normal charging flowpath (Step 25 of EP-3).

0706** "A" and "B" pressurizer spray valves closed.

0709 Alarm Typewriter: Non Regenerative Heat Exchanger outlet flow 47 gpm. (Evaluation of this entry indicates that normal letdown had been restored in accordance with Step 29 of EP-3.

0710 Alarm Typewriter: Pressurizer heater breakers closed (Step 31 of EP-3).

0711 Alarm Typewriter: Presrurizer heater breakers closed (Step 31 of EP-3).

0713 Alarm Typewriter: Secured "C" and "B" RCP's (Step 38 of EP-3).

0714 Alarm Typewriter: Spray demand 76%. (From this time forward RCS pressure is maintained by manual control of spray and heaters.)

0715 Superintendent of Operation and SRO-On-Call arrived in the Control Room.

0718 Transitioned to ES 3.1 "POST-STEAM GENERATOR TUBE RUPTURE COOLDOWN USING BACKFILL" (Step 42 of EP-3).

0720 Station Manager arrived in the Control Room.

0721 Alarm Typewriter: AFW Feed Pump 3A to "C" Steam Generator secured.

0722 Alarm Typewriter: Pressurizer level 73% and decreasing.

0723 Alarm Typewriter: AFW Feed Pump 3B to "B" Steam Generator secured. (Subsequently AFW pumps are run intermittently to support Steam Generator feed requirements.)

0725 Alarm Typewriter: Started "B" Condensate pump (Both "A" and "B" Condensate pumps now running).

0727** Began RCS cooldown in accordance with ES 3.1.

0730 Assistant Station Manager arrives in Control Room and initiates transition of EPIPs and communications from Control Room to TSC.

0739 Station Manager assumes Station Emergency Manager position.

0745 Alarm Typewriter: Turbine on the turning gear.

0756 Condenser Air Ejector manually diverted to containment.

0757 Technical Support Center activated.

0810 Alarm Typewriter: Secured "B" condensate pump.

0820 Corporate Emergency Response Center activated.

0845 Started "B" RHR pump for system warm-up (Step 9 of ES 3.1).

0853 Alarm Typewriter: Closed "A" MFP breakers. (Breakers in test to permit opening of pump discharge valve to use Condensate Pumps for feed to Steam Generators.)

0857 Alarm Typewriter: Open "B" MFP Breakers (To permit isolation of "B" MFP to stop spraying from "B" pump suction relief valve.

0900* Loose Parts Monitoring System Alarm on "C" Steam Generator.

0915 Local Emergency Offsite Facility activated. Commenced using auxiliary spray to supplement RCS depressurization.

0949 Containment partial pressure exceeded allowable set point due to Air Ejector exhaust diversion to containment.

1049 Pressurizer PORV Key Switches to "AUTO" for NDTT protection. (SRO Log)

1108 Entered Mode 4.

1153 Cycled reactor trip breaker to re-enable automatic Safety Injection.

1200 Placed "A" and "B" charging pumps and "B" LHSI pump in "Pull-to-Lock" in accordance with 1-OP-3.3.

1219 Placed RHR System in service to continue RCS cooldown (Step 9 of ES 3.1).

1221 Secured "A" Reactor Coolant Pump. (SRO Log)

1254 Main Steam System secured in accordance with 1-OP-28.1.

1312 Restored Air Ejector exhaust to normal alignment.

1330 Entered Mode 5

1335 Station Emergency Manager terminated the emergency.

1336 Notified Nuclear Regulatory Commission, State and Local Governments of termination of emergency status.

1336 Implemented Recovery Organization.

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*Approximate time based on CRO, SRO, and/or STA observation.
**Approximate time based on computer or strip chart data.

07/22/87

PRELIMINARY EVALUATION
NORTH ANNA UNIT I, C S/G TUBE FAILURE
AT SUPPORT PLATE 7, COLD LEG

An evaluation was made of the video tapes generated by Westinghouse, using fiber optics, of the R9-C51 tube failure. The following observations were made:

1. The tube failed over 360 degrees of circumference and the severed ends displaced in the axial direction approximately ½". The tube failed just above support plate #7 on the cold leg side.

2. As viewed from the cold leg side upward at the break location, an area of 60 degress or less is noted to be angled to the tube O.D. This may represent a final failure location in tensile overload or cyclic bending.

3. The fracture surface is generally rough and granular in appearance.

4. As viewed from the side, from the tube ID, the fracture edge is irregular and appears to be circumferential in orientation with little or no axial orientation of the elements of the crack.

5. Where several small axial cracks, or tears, do appear, they seem to be associated with a small thin zone of final rupture. They do not appear to be individual axial cracks.

6. The fracture surface does not appear to show a zone of flat fracture which might be associated in an initial fatigue crack. Although some cyclic bending may have been associated with the final rupture, no indication of fatigue is obvious as a possible crack initiation point.

7. The rough irregular nature of the edge of the fracture is similar to edge features produced by stress corrosion cracking.

8. There is no clear indication that the fracture initiated from the ID rather than the OD. The outside OD edge of the fracture cannot be viewed by the fiber optics probe.

9. There are indications from the video tape that what may be a small parallel zone of irregular circumferential cracking is visible in the 90 degree angle tape. This small zone of cracking appears to be just below the primary fracture which would place it at the level of the top of the No. 7 support plate.

10. The use of video tapes, without further laboratory work, is not considered sufficient to clearly identify the cause and nature of the failure.

J. M. McAvoy

Attachment 3

INSPECTION AND REPAIR HISTORY

•Unit Start-up in 1978

•1979 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 440	None	S/G A - 94
S/G B - 139	None	S/G B - 94
S/G C - 480	2 leaks in S/G C	S/G C - 96

Comments: Resin intrusion during cycle. Row 1's preventively plugged. 2 other tubes plugged due to denting. Denting first observed, Boric acid treatment initiated. Leakage rate barely detectable.

•1982 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 107	None	None
S/G B - 1165	None	None
S/G C - 243	None	None

Comments: Partial tube end repair due to split pin damage in S/G's A and C.

•1984 Forced Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G B - 579	3 leaks in S/G B	S/G B - 4
S/G C - 552	2 leaks in S/G C	S/G C - 5

Comments: No progression in tube denting observed. Row 1 leaking explosive plugs repaired. Partial tube end repair performed. Distorted indications at support plates first noticed. Leakage rate 396 GPD.

•1984 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 100% available	None	S/G A - 10
S/G B - 100% available		S/G B - 1
S/G C - 100% available		S/G C - 5

Profilometry in all 3 S/G's.

Comments: Partial tube end repair performed. Attempted tube removal in A S/G. Distorted indications observed. Foreign object located and removed in S/G C. 2 tubes plugged preventively. Leakage rates 2.3 GPD in A, and 10.8 GPD in C.

some ID degradation; also ID cracking at U-bends
28 Row 2 tubes plugged

•1985 Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 830	3 leaking tubes	S/G A - 13

Comments: Distorted indications observed. Leakage rate 213 GPD.

•1985 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 100% available*	None	S/G A - 9
S/G B - 100% available*	2 leaks in B	S/G B - 17
S/G C - 100% available*	4 leaks in C	S/G C - 47

Comments: Two tubes removed with 4 support plate intersections. 30 tubes from the three steam generators were plugged due to "strong" distorted indications. Sample specialized NDE applied in S/G C. Leakage rate 90 GPD.

* 100% available = all but plugged tubes

•Other Events During 1986 thru March 1987

- Extensive examination of tubing and materials with EPRI and Westinghouse.
- Preparation and submission of WCAP to NRC.
- Requested and held meeting with NRC staff in March, 1987.
- Developed eddy current rule base for April 1987 Refueling.

•1987 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 100% available*	None	S/G A - 83
S/G B - 100% available*	2 tubes in B	S/G B - 62
S/G C - 100% available*	4 tubes in C	S/G C - 118

Comments: Extensive additional NDE performed included:

- Profilometry of more than 100 tubes in each S/G.
- 8 X 1 probing of nearly 100% of available tubes. [illegible]
- Rotating pancake probing of all identified tubesheet indications and a sample of support plate intersections.
- AVB indications first noted, primarily in B S/G. All indications less than 40% and no tubes plugged.

Tube end repair completed. U-bend stress relief performed on all available Row 2 tubes in all 3 steam generators. Support plate stress relief demonstration performed in S/G B. Two tubes removed from S/G A containing 2 tubesheet indications and one support plate intersection. Leakage rate: 11.5 GPD in B and 14.6 GPD in C.

Tubes expanded through full length of tube sheet.
may extend
[illegible] ID [illegible] at tube support, above or below.
caused by [illegible]? more at [illegible]

* 100% = all but plugged tubes

NORTH ANNA UNIT 1
TUBE PLUGGING SUMMARY

OUTAGE DATE	STEAM GENERATOR A	B	C	TOTAL TUBES
SEPTEMBER '79	94	94	96	284
JANUARY '84	0	4	5	9
MAY '84	10	1	5	16
AUGUST '85	13	0	0	13
NOVEMBER '85	9	17	47	73
APRIL '87	83	62	118	263
TOTAL	209 (6.2%)	178 (5.3%)	271 (8.0%)	658 (6.5%)

ATTACHMENT 3 (Cont)

Revised
07/21/87

STEAM GENERATOR INSPECTION AND MAINTENANCE
"C" STEAM GENERATOR
1987 REFUELING OUTAGE

- Eddy Current Inspection:

  - 516 tubes inspected full length (16%)

  - 247 tubes inspected through the hot leg to the #7 support plate, cold leg side (7.6%).

  - 2472 tubes inspected through the #7 support plate, hot leg side (76.4%).

  - All available tubes inspected using 8 x 1. Inspection encompassed all available tubes on the hot leg side (tube sheet area).

  - RPC inspection performed on 41 tubes at top of tubesheet, hot leg side.

  - Plugging of 118 tubes due to support plate or tubesheet indications. One (1) tube out of total plugged due to error. (No indication in tube).

  - Profilometry inspection of 121 tubes through the #7 TSP Hot leg.

- Other Maintenance and Inspection Activities:

  - Row 2 U-Bend Stress Relief. (75 returned to service).

  - Inspection of J-tubes (8 sampled). Also, visual examination of steam drum.

  - Sludge Lance. Thirty (30) passes removed 1610 pounds of sludge.

  - Annulus Inspection of steam generator. Both hot and cold leg side.

  - Flowslot photography

  - * - Removed and re-installed tube lane blocking devices.

Dos line appears to be [illegible]

Attachment 3 (cont)

'C' S/G DATA SUMMARY
AS OF 7/24/87

The IS sample selected for the 'C' S/G inspection is based on the following:

- Satisfy IS T.S. sample plan
- Sample shall include:
  1) all previously identified degraded tubes (degraded defined as any callable indication)
  2) tubes identified by 3x3 grid for rows 10-46 and a 3x4 grid for rows 2-9 (tube will be excluded if previously plugged)
  3) the 8 tubes surrounding the failed tube

To date the standard bobbin coil inspection has been performed from the hot leg on a total of 366 tubes (Westinghouse analysis is complete on all 366) out of a total of 374. Tubes in rows 10-46 were inspected from tubesheet to tubesheet. Rows 2-9 were inspected to the 7th support plate on the cold leg side. Of the 366 tubes analyzed there have been five distorted indictations (DI's) identified and one clear indication. The following summarizes these indications and provides a review of the spring refueling outage data for these tubes.

Row	Column	Spring Data	July Data	Explanation
16	10	Not identified	DI	DI is located at the 6th support plate on the hot leg. Indication was missed in spring inspection. Signal appears the same now as in spring.
9	32	Not tested	DI	DI indication just above the 7th support plate on the cold leg. This area was not inspected during the spring outage.
31	49	Not identified	78%	Indication is located approx. 1/2 in. above the tubesheet. Indication was missed in spring outage.
19	19	No flaw apparent	DI	DI located just above the 6th support plate on the cold leg. Signal appears to have changed.
34	49	No flaw apparent	DI	DI located just above the 1st support plate on the hot leg. Signal appears to have changed.
25	58	No flaw apparent	DI	DI located just below the 2nd support plate on the hot leg. Signal appears to have changed.

In addition to the standard bobbin coil inspection, an 8x1 inspection has begun on 'C' S/G on the hot leg side to just past the 7th support plate. The initial 8x1 inspection plan consisted of 150 tubes in the columns around column 51. Of the tubes inspected (107), 19 have been analyzed by Westinghouse. The results of these analysis show two possible indications. These indications have not been verified with RPC. Neither of these tubes were inspected beyond the hot leg tubesheet region during the spring refueling outage.
The indications are summarized below:

Row	Column	Indication Location
46	49	3rd and 4th support plate hot leg
46	50	1st support plate hot leg

ATTACHMENT 4

PROBE TYPES USED FOR STEAM GENERATOR TESTS

DIFFERENTIAL BOBBIN PROBE

Coils are coaxial with the tube, about 0.050 in. long and about 0.050 in. apart. They are usually 0.720 in. in diameter and are operated in an absolute and differential bridge mode. With the MIZ18 eddy-current instrument, they are driven at four multiplexed frequencies (10, 200, 400, and 600 kHz). The eddy-current pattern in the tube is also coaxial to the tube, and any tube property that interrupts or changes the flow of eddy currents will cause a change in the coil impedance. These tube property variations include tubesheets, tube supports, dents, magnetite on the tube or in the crevice, defects in the tube, and intergranular attack. Only the axial component of defects will interrupt the circumferential flow of eddy currents produced by the bobbin coil so that circumferential defects, with very little axial component, produce very low amplitude signals. These signals can be easily lost among signals from other property variations.

8 x 1 PROBE

This probe consists of eight independent pancake coils operated in an absolute mode, being driven at 200 and 400 kHz. These probes are typically 3/16 in. in diameter, one in. long, and contoured to fit the curvature of the tube. The eight coils are arranged in two rings of four coils each, and overlapped in a manner such that every point on the tube passes under at least one coil. Each coil is individually spring loaded against the tube to minimize distance between the coil and tube wall, or "lift-off." The eddy-current flow pattern from these coils is circular, around the coil axis, and a crack of any orientation will interrupt the main flow of eddy currents. The coil is smaller than the bobbin coil and has a more concentrated field, so a small defect causes a larger change in signal. The coil is, however, more sensitive to the variations in coil-to-conductor spacing or lift-off than the larger bobbin coil. While the spring loading against the tube wall helps, irregular and sharp dents will give a substantial lift-off signal. Since information at only two frequencies are recorded (200 and 400 kHz), this coil type does not have as much data available as the bobbin or rotating pancake coil.

ROTATING PANCAKE COIL (RPC)

This probe is similar to the individual 8 x 1 coils, but is smaller (typically 0.125 in.). It has a still smaller focus, which gives better resolution to small defects, sees less of the tube outer diameter artifacts, and is more sensitive to lift-off. The probe head, containing the coil, is rotated and the coil is sprung against the tube wall. Data are recorded at three frequencies (at least), and a very fine and time-consuming scan is made of a "suspected area" of a tube. The spring loading and size of this probe are such that it rides the surface fairly well, and a three-dimensional plot of the data gives a good contour of any defects.

ATTACHMENT 5

1600 7/22 TO 0800 7/23 Inspection Plan

1. Complete 15 sample on hot leg.

2. Perform endoscope inspection from hot leg.

3. Start initial 8 x 1 inspection   Hot leg side to U-bend
   Rows 2-12 Column 48
   Entire columns 49-51
   Rows 3-13 Column 52

4. Verify 8 x 1 data with RPC as needed.

BEYOND 0800 7/23

1. 100% 8 x 1 Hot leg through the 7th support plate.

2. RPC verification of 8 x 1 indications.

3. Profilometry of verified indications.

4. Retest as required.

5. Plug as required.

6. Remove SM-10 fixture.

7. Set-up in cold leg.

8. Complete 15 inspection.

9. Perform standard bobbin on portions not inspected in spring outage

10. 100% 8 x 1 inspection cold leg through 7th support plate.

11. RPC and profilometry verification as required.

12. Plug as required.

13. Remove SM-10 fixture.

A TACHMENT 6

STEAM GENERATOR INSPECTION STATUS

DATE 08/07/87

	Standard Bobbin	8x1	RPC	Profilometry
Total to Insp.				
A	2685	3179	11	TBD
B	2662	3210	9	TBD
C	374(H) 2390(C)	3117	36	50
No. Inspected				
A	2685	589(C)	0	-----
B	2667	1018(C)	0	-----
	370(H)	1566(H) 604(C)	50	-----
No. Analyzed by W				
A	790	--0----	-----	-----
B	1703	0	-----	-----
C	370(H) 0(C)	104(H) 11(C)	11	50

S/G	DI's	8x1 PI's	Cleared By RPC	Verified By RPC	Clear Indications	Number To Be Plugged
A	11	-----	------	-------	0	------
B	9	-----	------	-------	0	------
C	21	15	10	1	1	------

8x1 testing is complete. All of first shift (total of 7) passed, others are being graded. RPC testing will be done on a limited basis (approximately 6).