SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO RESTART AND OPERATION AT 50% POWER
FACILITY OPERATING LICENSE NO. NPF-4
VIRGINIA ELECTRIC AND POWER COMPANY
OLD DOMINION ELECTRIC COOPERATIVE
NORTH ANNA POWER STATION, UNIT NO. 1
DOCKET NO. 50-338

INTRODUCTION

By letter dated September 22, 1987, the Virginia Electric and Power Company (the licensee) requested that the North Anna Power Station, Unit No. 1 (NA-1) be permitted to start up and operate at 50 percent of full power. Following the July 15, 1987 NA-1 Steam Generator Tube Rupture (SGTR) event, the licensee agreed to obtain concurrence from the NRC prior to NA-1 restart (Mode 2). This agreement was specified in the NRC Confirmatory Action Letter (CAL) issued July 22, 1987. The licensee has completed the evaluation of the SGTR event and has submitted by letters dated September 15 and 25, 1987 the evaluation of the SGTR event, including the SGTR failure mechanism and modifications to be made prior to restart. The NRC review of these matters may extend beyond early October, 1987 when NA-1 is scheduled to be ready for restart. Therefore, as noted above, the licensee has requested NRC concurrence for restart and operation of NA-1 at 50 percent of full power pending NRC authorization for full power operations. In order to place these matters in proper perspective, a brief description of the NA-1 SGTR event and the licensee's investigation of this event is provided below.

Prior to 0630 hours on July 15, 1987, NA-1 was operating at 100% power. At 0630 hours, the Main Steam Line "C" radiation monitor registered a Hi-Hi alarm and the Control Room Operator (CRO) noted pressurizer (PZR) level and pressure decreasing. Therefore, the CRO increased charging flow to the Reactor Coolant System (RCS). The unit was manually tripped at 0635 hours and approximately 20 seconds later a Lo-Lo pressure safety injection signal actuated automatic trip. At 0639 hours a Notification of Unusual Event was declared and at 0650 hours feedwater flow to SG "C" was isolated. However, the level of SG "C" was identified to be increasing, indicating an SG tube rupture or break (SGTR). At 0654 hours an alert was declared and at 0705 hours Safety Injection (SI) was terminated. At 0710 hours emergency procedures were initiated for post-SGTR cooldown using backfill. The Technical Support Center was activated at 0757 hours and the local emergency operations facility activated at 0915 hours. The unit entered Mode 4 (Hot Shutdown) at 1108 hours and at 1218 hours the RHR system was placed in service. The unit entered Mode 5 (Cold Shutdown) at 1330 hours and the event was terminated at 1335 hours.

No automatic actuation of primary or secondary safety relief valves occurred. Total radioactivity release was less than 1% of Technical Specification (TS) limits. The tube leakage rate (as determined later) was in the range of 560-637 gallons per minute (GPM). Offsite environmental monitoring teams detected no increase in radioactivity above normal background levels. The SGTR event was determined to be bounded by the Updated Final Safety Evaluation Report (UFSAR). The maximum leak rate (560-637 gpm) was less than the UFSAR value of 710 gpm. Core safety limits were not challenged and shutdown and thermal margins were maintained.

Once access to the NA-1 SGs A, B, and C was gained, the licensee's evaluation of the SGTR event was oriented to: (1) Determining the root cause of the failure; (2) Ascertaining the condition of the SGs, particularly with respect to the failure mechanism; and (3) Performing the necessary corrective actions to preclude the future occurrence of a tube rupture event.

On July 21, 1987, VEPCO identified a ruptured tube in SG-C. The tube location was Row 9, Column C51 (R9 C51) on the cold leg side at the seventh support level. A fiber optic examination identified the failed tube to be the classical double-ended guillotine break. On August 12, 1987, VEPCO successfully completed the removal of tube R9 C51 on the cold leg side up to and including the break at the seventh support level. The tube was immediately sent to Westinghouse for an extensive nondestructive/destructive examination to determine the fracture morphology and the failure propagation mechanism. The results of these examinations and the determination of the tube failure mechanism are provided in the licensee's final report dated September 15, 1987, and are discussed below.

In order to provide justification for future restart of NA-1, VEPCO has conducted an extensive inspection of all three SGs. The inspection has been the most extensive eddy-current testing program undertaken at a U.S. domestic facility with emphasis on detecting circumferential defects.

Eddy current testing (ECT) is the principal method used for performing tube inspections. This inspection method involves the insertion of a test coil inside the tube that traverses the tube length. The test coil is excited by an alternating current, which creates a magnetic field that induces eddy currents in the tube wall. Disturbances of the eddy currents caused by flaws in the tube wall produces corresponding changes in the electrical impedance as seen at the test coil terminals. Instruments are used to translate these changes in test coil impedance into output voltages which can be monitored by the test operator. The depth of the flaw can be determined by the observed phase angle response. The test equipment is calibrated using tube specimens containing artificially induced flaws of known depth.

The ECT testing program has included the inspection of every tube support junction and straight tube sections in all three SGs with an 8x1 pancake array probe. This probe (8x1) has the sensitivity to detect all inner-diameter defects, either axial or circumferential, and with defects 20% or deeper and with a length of 3/16 of an inch or longer. Also, the 8x1 probe is able to detect outer-diameter cracks and intergranular attack on either the inner-or outer-diameter. In addition, all indications detected by the 8x1 probe have been tested with the Rotating Pancake (RPC) probe. Finally, profilometry has been conducted on selected intersections.

A Westinghouse Intelligent Eddy Current Data Analysis System (IEDA) has been used as an aid in flagging suspect bobin coil indications, which are then dispositioned by data analysts. The data from each tube has been independently reviewed by two different analysts. One analyst has used the Westinghouse IEDA system and the other analyst has used the Zetec Digital Data Analysis System.

All data analysts are certified at least Level II in accordance with American Society of Nondestructive Testing (ASNT) requirements. The analysts have been given additional training by Westinghouse and required to pass a test that covers the specific data analysis being used for the present NA-1 eddy current tests.

Finally, it is noted that an NRC Augmented Inspection Team (AIT) was dispatched to the NA facility. The AIT was charged with determining whether the licensee's actions in response to the July 15, 1987 SGTR were adequate to protect the health and safety of the public and that appropriate action was being initiated to determine the cause of the event. In addition, the procedures followed by the licensee relative to the SGTR were evaluated to assess the adequacy of in-place procedures to cope with serious events of this type.

The NRC AIT Report was issued August 28, 1987. The Report, in part, concluded that, "The overall results achieved were outstanding in that the operator tripped the plant, isolated the leak and brought the plant to cold shutdown in seven hours without using the S/G power-operated relief valves. This contributed to a negligible release to the environment."

Our discussion and evaluation of these matters with respect to restart of NA-1 for operations not to exceed 50% of full power are provided below.

DISCUSSION

Steam Generator Inspection

As noted above, the licensee conducted an extensive SG ECT inspection of the NA-1 SGs A, B and C. Identified indications were either present in the April 1987 refueling outage with no discernable change indicated or in previously uninspected portions of each SG. Additionally, a review of the data from the last outage using the present analysis rules revealed several tubes that should have been plugged at the previous outage. This apparent, though not actual, change in the SG condition is due to the change in the analysis rules and increased awareness by the analysts of North Anna specific ECT signals. A review and comparison of the SG C hot leg data demonstrates that there is essentially no change in tube condition from the April 1987 refueling outage to July 1987 (when the event occurred). Of significant importance was the fact that there were no indications of circumferential nature found at any tube support plate locations, including the seventh tube support plate.

The number of tubes inspected is shown below. Each steam generator contains 3388 tubes. However, a number of tubes have been plugged from previous SG inspections. The number of non-plugged tubes are: SG A - 3179; SG B - 3210; and SG C - 3117.

The number of tubes to be removed from service based on the SGT inspection by indication type are indicated in the following table.

STEAM GENERATOR TUBES TO BE PLUGGED AS RESULT OF SGTR EVENT
(By Indication Type)

S/G	Clear[1] Indications	Distorted[2] Indications	Tube[3] Sheet Indications	8x1 Possible[4] Indications	Other[5]	Total[6] tubes to be Plugged
A	0	6	6	11	2	25
B	0	3	5	12	1	21
C	2	2	20	11	4	39

[1]Clear Indications (defective) - bobbin indications of greater than 40 percent "thru-wall" depth.

[2]Distorted Indications - bobbin indications of undetermined "thru-wall" depth at tube support plates.

[3]Tubesheet indications - bobbin indications of undetermined "thru-wall" depth at tubesheet.

[4]8x1 Possible Indications - indications identified by 8x1 probe.

[5]Tubes with broken probes or which would not pass 8x1 probe - includes failed tube.

[6]Plugging summary is as of 9/14/87 based on ECT results - does not include tubes to be plugged as a preventative measure based on fatigue considerations or other concerns.

Tube Failure Mechanism

Upon arrival at Westinghouse, tube R9 C51 (the tube with the circumferential break) was immediately subject to a series of non-destructive/destructive tests to determine the tube failure mechanism. Visual examinations and macroscopic examinations of the tube fracture surface were conducted to determine crack origins and crack propagation paths. Scanning Electron Microscope (SEM) and Transmission Electron Microscope (TEM) fractographic examinations were also performed to confirm tube crack origins and crack propagation paths.

Mechanical properties of the tube were determined and found to agree closely with the 1971 tube certification data applicable to NA-1. Microstructure was typical of mill-annealed Alloy 600 for NA-1. Grain size was small, ASTM 9.5.

Based on the above, the cause of the failure was determined to be fatigue. No evidence of any significant intergranular corrosion was observed on or immediately adjacent to the fracture surfaces. High cycle fatigue striations were present and were measured to obtain the stress intensity which led to initiation of the fatigue crack and crack propagation. The mode of crack propagation concluded that leakage occurred between the time of total through-wall development of the crack front and the final circumferential break.

The orientation and spacing of the striations support the conclusion that normal design operational loadings were not sufficient to lead to the fatigue failure. Therefore, some other loading mechanism was acting on the tube to produce the failure. Measurements of the striation spacing provided necessary data to determine the range of loadings that led to eventual fatigue of the tube. Adverse flow mechanisms were evaluated, such as turbulence, vortex shedding, and fluid elastic excitation. Review of the data supports the conclusion that fluid elastic excitation was the most probable mechanism that could provide sufficient loadings or alternating stresses to induce fatigue.

An additional method was utilized to determine these loadings and verify the striation spacing measurements and resultant loading conditions. This method used tube dent data (obtained through profilometry and physical measurements) and finite element analysis to establish mean stress data through the dent. This mean stress data, the dented configuration and fatigue curve were then used to determine the alternating stress intensity required to initiate a fatigue crack. This calculated range of stress intensity supported the similar conclusion determined from striation spacing measurements that tube failure was induced by fatigue.

A fluid elastic stability ratio was defined for failed tube R9 C51. The stability ratio represents a measure of the potential for tube vibration due to instability during service. Values greater than unity (1.0) indicate fluid elastic instability. The fluid elastic stability ratio is defined as the effective velocity divided by the critical velocity. The calculated flow ratio was determined for current NA-1 flow parameters. Calculations determined that the tube would be more susceptible to fluid elastic instability due to lower damping caused by denting. Simulated shaker tests supported the conclusion that in this regime of low damping, tube R9 C51 would be fluid elastically unstable.

As discussed above, the results of the present SG inspection indicated no eddy current indications of a circumferential nature at any seventh support plate location. This is consistent with the fatigue mechanism described above. The majority of the fatigue process lies in the cyclic loading (via alternating stress) to initiate a crack (or cracks) in the tube. Once the fatigue crack initiates, the time required to propagate the crack is comparatively small.

Antivibration Bars (AVBs) limit the high vibration amplitudes needed to achieve the alternating stress necessary for fatigue crack initiation. The depths of AVB penetration into the SG tube bundle can be estimated from eddy current indications that can then be translated to a SG inspection map which provides an indication of non-uniform AVB insertion depths.

A large number of AVB indications were identified during the current SG inspection. This is not unusual in a Series 51 Westinghouse SG. However, a few indications were identified as far down as Row 8. Therefore, extensive eddy current testing was performed to identify AVB indications. The inspection revealed that the majority of the Row 9, 10 and 11 tubes were supported by AVBs. However, failed tube R9 C51 was not supported by an AVB.

Correlation with the known deflections required to provide sufficient stress to initiate fatigue show that the AVBs limit the tube motion to below the required deflection limit. This data provided further support to the conclusion that the loading mechanism for R9 C51 was fluid elastic excitation.

In summary, the licensee concludes that the tube failure was due to high cycle fatigue. The fatigue mechanism was determined to be a combination of stresses imposed by tube denting at the seventh support plate and vibration due to fluid elastic instability.

Corrective Actions

The licensee has implemented a series of corrective actions and modifications to preclude similar tube failures at NA-1. These include the installation of a downcomer flow resistance plate in order to reduce the loadings experienced by susceptible tubes. Preventive SG tube plugging is being implemented to further reduce the probability of tube rupture. In addition, an enhanced monitoring program is being implemented to provide sufficient notification of tube leakage in order to shut down NA-1 prior to a tube rupture. These matters are discussed below.

(1) Downcomer Flow Resistance Plates

The NA-1 SGs A, B and C are being modified to include a downcomer flow resistance plate (DFRP). The DFRP will reduce the steam generator recirculation flow and is expected to result in the improvement in tube "stability ratio" needed to preclude further tube failures of active tubes by the fluid elastic instability mechanism. As noted above, "stability ratio" is a relative measure of the potential for tube vibration due to fluid elastic instability. Evaluations by the licensee have concluded that a 10% improvement in stability ratio should provide the necessary reduction in fatigue usage (reduced amplitude of vibration) to preclude further tube failures by this mechanism over the remaining life of the steam generators. The installation of the DFRPs will be completed prior to NA-1 restart.

For operations at greater than 59 percent power, the Final Safety Analysis Report (FSAR) will be revised to include the DFRP (reduced mass flow) in the SGTR accident analysis. A reanalysis of the SGTR event with the DFRP has resulted in a calculated offsite dose which is greater than reported in the UFSAR. The increase in dose consequences for the SGTR event occurs only for rated thermal power levels above approximately 59%, and the consequences are still well within established acceptance criteria as defined in the UFSAR and the bases for NA-1 TS 3/4.4.8.

(2) Preventive Plugging

Preventive plugging will take place on the potentially susceptible tubes in Row 8 through 11. The essential criterion for identifying specific tubes for preventive plugging is that they not be supported by at least one AVB. All such tubes will be plugged. On the cold leg side, each tube meeting this plugging criteria will be plugged with a sentinel plug. The sentinel plug will permit internal pressurization of the tube and low level leakage in the event a through-wall crack develops in the plugged tube. This will serve as an early warning detection method for occurrence of a similar circumferential break of a plugged tube, thus precluding impact with any adjacent tubes which are in service. This plugging will be completed prior to restart fo NA-1.

It is noted that the combined effect of the DFRP and the preventive plugging discussed above represents a reduction of 24 percent in the stability ratio referred to above.

(3) Stabilization of Failed Tube R9 C51

The failed tube R9 C51 will be stabilized to prevent further damage to adjacent tubes. In addition, its neighboring tubes will be preventatively plugged with sentinel plugs.

(4) Tube Plugging-Eddy Current Testing

Based on the extensive SG eddy current testing program, all tubes having ECT indications will be plugged to meet established plugging criteria. These tubes will be plugged prior to restart for NA-1.

(5) Augmented Surveillance Program

An augmented surveillance program for monitoring steam generator primary-to-secondary leakage is being implemented. This program is based on the use of several radiation monitors and sampling (including the installation of a new N^{16} gamma detection system) to quantify primary-to-secondary leakage. The program is designed to detect leakage during the early stages of fatigue failure so that an orderly shutdown can be accomplished. Administrative controls addressing leak rate limits, operator actions and monitoring equipment operability are being prepared. The N^{16} gamma detection system will become operational prior to power ascension greater than 30 percent of full power. All other aspects of surveillance program will be in place prior to restart for NA-1.

STABILITY RATIO AT 50 PERCENT LOAD (POWER)

The steam flow rate from a steam generator is highly dependent on thermal load. As the load decreases the steam flow decreases. In addition, a decrease in load is accompanied by a decrease in void fraction. Both of these effects work to reduce the tube stability ratios. At 50% load, the effect on stability ratios from steam flow (velocity and density) is a 26% reduction in stability ratios relative to full load and prior to any of the modifications noted above. In addition at 50% load there is a reduction in void fraction of about 10% which increases tube damping. This reduction in void fraction, if considered, would result in a further reduction in stability ratios over the 26% noted above.

Studies performed for the licensee by Westinghouse have determined that a SG tube similar to R9 C51 requires a 10% reduction in the stability ratio in order to reduce the fatigue usage factor (cycles to failure per year) to a value substantially less than unity (1.0) for the remaining design life of the SGs. Thus, at 50% load, the required reduction in stability ratio is achieved with at least a factor of 2.6 for safety. Therefore, it is highly unlikely that a fatigue crack would initiate due to this mechanism at 50% load and prior to any modifications.

In addition, the licensee is modifying the steam generators to install DFRP's and preventively plug certain tubes which are considered most susceptible to fluid elastic instability. The licensee has determined that the net effect of operation at 50% load in combination with the DFRP and preventative plugging represents a reduction in stability ratios of 60% relative to stability ratios at full load and prior to any modifications. Thus, at 50% load, the required reduction in stability ratio is achieved with a factor of 6.0 for safety. Finally, the licensee has determined that at 50% load, and with these modifications, no tube remaining in service will have a stability ratio greater than unity. This indicates that the tubes will be stable and that a fatigue crack cannot initiate due to the fluid elastic instability mechanism at 50% load.

SUMMARY

The staff concurs with the licensee that the cause of the SG C, R9 C51 tube failure was fatigue. The staff is continuing to evaluate the exact mechanism of the tube failure and the licensee's overall program of multiple actions designed to prevent a similar tube rupture event. However, the staff finds there is reasonable assurance that the adverse fluid flow conditions which led to excessive tube vibration that caused the rapid propagation fracture will not be present at 50 percent power operation. Therefore, interim operation at reduced power (less than or equal to 50 percent power) is acceptable.

The staff's final evaluation of the tube failure mechanism and multiple actions designed to prevent a similar tube rupture will be completed and issued prior to any authorization for operation at greater than 50% power.

Implementation and Verification of Modifications Prior To Restart (Mode 2):

Prior to restart of NA-1 (Mode 2), NRC, Region II will verify that the following modifications and procedures have been completed or followed as specified below:

(1) The adequacy of the licensee's operating procedures for SG leakage rate surveillance,

(2) That SG tube R9 C51 has been stabilized in conformance with vendor (W) recommendations

(3) That flow restrictor plates have been installed in conformance with vendor (W) recommendations, and

(4) That applicable procedures have been followed for loose parts accountability.

In addition, prior to any power ascension greater than 30 percent, the operability of the newly installed N^{16} monitor shall be verified to be operable.

Dated: October :, 1987

Principal Contributor:

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