REQUEST TO CHANGE PROCEDURE
NORTH ANNA POWER STATION
VIRGINIA POWER

TO ..PERVISOR RESPONSIBLE FOR FOLLOWING PROCEDURE: 1

- ☐ ABNORMAL
- ☐ ADMINISTRATIVE
- ☐ ANNUNCIATOR
- ☐ CALIBRATION
- ☐ CHEMISTRY
- ☐ CURVE BOOK
- ☐ EMERGENCY
- ☐ IN-SERVICE INSPECTION
- ☐ MAINTENANCE
- ☐ NON-DESTRUCTIVE TEST
- ☐ OPERATING
- ☐ PERIODIC TEST
- ☐ HEALTH PHYSICS
- ☐ SPECIAL TEST
- ☐ START-UP TEST
- ☐ WELDING
- ☐ ______
- ☐ ______
- ☐ ______
- ☐ ______

PROCEDURE NO: 1-OP-2.2 2 UNIT NO: 1 3 REVISION DATE: 9-4-86 4

TITLE: UNIT POWER OPERATION MODE 1 TO MODE 2 5

CHANGES REQUESTED: (GIVE STEP NUMBER, EXACT SUGGESTED WORDING, AND LIST REFERENCES, STAPLE COPY OF PROCEDURE WITH SUGGESTED CHANGES MARKED TO THIS FORM.) 6

ADD STEP 4.3 & 4.10.1 TO NOTIFY STA TO PLACE 50 (30) % CORRECTION FACTORS INTO THE N-16 RAD MONITOR MICROPROCESSOR

REFERENCES:

CHUR - 87 - 569

REASON FOR CHANGES: 7

TO INSURE N-16 RAD MONITOR IS INDICATING ACCURATELY AND IN THE CONSERVATIVE DIRECTION.

CH. . REQUESTED BY: [signature] 8 DATE: 10-19-87 9

ACTION TAKEN: 10

DOES THIS CHANGE THE OPERATING METHODS AS DESCRIBED IN THE UFSAR✓ ☐ YES ☒ NO
DOES THIS CHANGE INVOLVE A CHANGE TO THE TECH. SPECS✓ ☐ YES ☒ NO
DOES THIS CHANGE INVOLVE A POSSIBLE UNREVIEWED SAFETY QUESTION✓ ☐ YES ☒ NO
IF ALL "NO", NO "SAFETY ANALYSIS" IS REQUIRED. IF ANY "YES", A "SAFETY ANALYSIS" IS REQUIRED. (10CFR50.59) APPROVED COPY TO BE PROVIDED TO LICENSING COORD. FOR INCLUSION IN ANNUAL REPORT.

RECOMMENDED ACTION: 11

☒ APPROVED ☐ DISAPPROVED

DOES THIS PROCEDURE CREATE A QA DOCUMENT✓ YES ☒ NO ☐

BY: (COGNIZANT SUPERVISOR) [signature] 12 DATE: 10-19-87 13

REVIEWED BY QUALITY ASSURANCE:

CHANGES MADE: YES ☐ NO ☒ 14

BY: [signature] 15 DATE 10/19/87 16

REVIEWED BY STATION NUCLEAR SAFETY AND OPERATING COMMITTEE: 17

☒ APPROVED ☐ DISAPPROVED ☐ APPROVED AS MODIFIED BY COMMITTEE

CHAIRMAN SIGNATURE: [signature] 18 DATE: 10/20/87 19

NEW PROCEDURE REVISION DATE: 20

ACTION COMPLETED BY: 21 DATE: 22

E/12

VIRGINIA POWER
NORTH ANNA POWER STATION
UNIT NO. 1

UNIT POWER OPERATION MODE 1 TO MODE 2

REFERENCES:

1. North Anna Unit 1 Technical Specifications
2. Westinghouse Operating Procedures
3. Westinghouse Precautions, Limitations and Setpoints
4. Westinghouse NSSS Manual
5. UFSAR

REV. NO.: 15 PAGE: ENTIRE DATE: 09-04-86 APPROVAL: ECS

RECOMMEND APPROVAL:

APPROVED BY:

CHAIRMAN STATION NUCLEAR SAFETY AND OPERATING COMMITTEE

SAFETY RELATED

DATE: 09-04-86

VIRGINIA POWER
NORTH ANNA POWER STATION
UNIT NO. 1

UNIT POWER OPERATION MODE 1 TO MODE 2

1.0 Purpose

1.1 This controlling procedure provides instructions for unit power operation from existing power level (Mode 1) to ≤ 5% power (Mode 2).

2.0 Initial Conditions

2.1 The unit is in Mode 1 with Reactor Power > 5%.

3.0 Precautions and Limitations

3.1 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the conditions of the limiting conditions for operation, as set forth in Technical Specifications, are met without reliance on provisions contained in the ACTION STATEMENTS. Unless otherwise excepted, this provision shall not prevent passage through OPERATIONAL MODES as required to comply with ACTION STATEMENTS.

3.2 T.S. 3.2.1 - The indicated Axial Flux Difference shall be maintained within a ± 5% target band (flux difference units) about the Target Flux Difference (for operation above 90% Rated Thermal Power).

NOTE: The flux must be within the target band within 15 minutes or the power must be reduced to below 90%.

3.3 The indicated Axial Flux Difference shall not be outside the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, for operation between 50% - 90% Rated Thermal Power.

NOTE: One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

NOTE: One-half minute penalty deviation for each one minute or POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

3.4 T.S. 3.2.4 - The QUADRANT POWER TILT RATIO shall not exceed 1.02 (for operation above 50% Rated Thermal Power).

3.5 IF S.R. Inst. is inoperable due to the inoperability of 2 or more P.R. Detectors, contact the Inst. Dept. to manually bias, jumper OR reinstate inoperable P.R. Detectors as required.

3.0 Precautions and Limitations (cont.)

3.6 Condenser Vacuum should be maintained as low as possible, ideally below 3.5" Hg, when the Unit is operating at low loads.

3.7 To prevent potential Turbine damage, the Unit should not be operated at low loads for extended periods without the MSRs in service.

Initials

4.0 Procedure

This procedure may be used to decrease power level from any power level ≤100% power to any power level ≤5% power as dictated by plant conditions. Mark all other steps NA, sign, date and route procedure noting reasons on the cover sheet. All steps between starting the power decrease and terminating the decrease must be signed off or a procedure deviation completed.

______ 4.1 Initial Conditions are noted and satisfied.

______ 4.2 Precautions and Limitations ~~are noted~~ have been reviewed.

NOTE: Auto rod control may be utilized from 15% to 90% provided that insertion limits are satisfied. Auto rod control may be utilized from 90% to 100% power provided "D" bank is greater than 215 steps.

______ 4.3 Notify the STA to place the 50% correction factors into the N-16 Rad Monitor microprocessor.

______ 4.~~3~~4 Commence unit load reduction utilizing main turbine "Reference Control" ∇ and GO pushbuttons, OR "Turbine Manual" ∇ as applicable.

NOTE: An Isotopic Analysis for Iodine, including I-131, I-133 and I-135 shall be performed between 2 and 6 hours following a thermal power change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

______ 4.~~4~~5 Borate, as per 1-OP-8.3, as necessary to maintain control rods in the normal operating range.

T.S.3.1.3.6 - The control banks shall be limited in physical insertion as shown in T.S. Figures 3.1-1 and 3.1-2.

______ 4.~~5~~6 Verify that the Mixed Bed Demineralizer is in service as per 1-OP-8.2 AND increase letdown rate to maximum, IF required.

______ 4.~~6~~7 As necessary, commence removing the high pressure drain AND low pressure drain systems from service as per 1-OP-34.0.

Initials

4.0 Procedure (cont.)

______ 4.~~7~~8 At approximately 70% Reactor Power transfer the Auxiliary Steam supply from 2nd point extraction to main steam (Unit 1 or 2) OR U-2 2nd point extration OR to the Auxiliary Boilers as per 1-OP-35.1.

NOTE: An Isotopic Analysis for Iodine, including I-131, I-133 and I-135 shall be performed between 2 and 6 hours following a thermal power change exceeding 15 percent of the RATED THERMAL POWER with a one hour period.

______ 4.~~8~~9 At approximately 55% Reactor Power verify that sufficient feedwater recirc. valves (FCV-FW-150A, B OR C) are open AND place their respective control switches to the "Open" position.

NOTE: An Isotopic Analysis for Iodine, including I-131, I-133 and I-135 shall be performed between 2 and 6 hours following a thermal power change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

NOTE: Pump operations are defined and limited by load shed system as per 1-OP-26.7.

______ 4.~~9~~10 At approximately 50% reactor power, one main feed pump may be secured, IF desired, as per 1-OP-31.1.

______ 4.10.1 NOTIFY the STA to place the 30% correction factors into the N-16 Rad Monitor microprocessor.

****

CAUTION: Condenser vacuum should be maintained as low as possible, ideally below 3.5" Hg, when the Unit is operating at low loads.

****

CAUTION: To prevent potential turbine Damage, the Unit should not be operated at low loads for extended periods without the MSRs in service.

****

4.~~10~~11 At 35% power, commence removal of the Reheat Steam System by: (NA this step (4.~~10~~11) IF the Unit is not going to be removed from service.)

Initials

4.0 Procedure (cont.)

4.~~10~~11.1 At the reheater control panel,

_______ Decrease valve positioner at zero

_______ Depress RESET button AND verify closed:

_______ FCV-MS-104A

_______ FCV-MS-104B

_______ FCV-MS-104C

_______ FCV-MS-104D

4.~~10~~11.2 Commence aligning the MSRs in preparation for start-up as per 1-OP-28.3.

_______ 4.~~11~~12 At 30% Reactor Power verify illumination of PR < 30% PWR P-8 PERM (Permissive Panel point D-4).

NOTE: An Isotopic Analysis for Iodine, including I-131, I-133 and I-135 shall be performed between 2 and 6 hours following a thermal power change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

4.~~12~~13 At 20% Main Turbine Power:

4.~~12~~13.1 Verify OR open ALL turbine drain valves:

_______ MOV-SD-100A

_______ MOV-SD-100B

_______ MOV-SD-100C

_______ MOV-SD-100D

_______ MOV-SD-101

_______ MOV-SD-102A

_______ MOV-SD-102B

_______ MOV-SD-102C

_______ MOV-SD-102D

NOTE: Pump operations are defined AND limited by load shed system as per 1-OP-26.7.

Initials

4.0 Procedure (cont.)

________ 4.~~12~~13.2 Remove one feedwater pump from service as per 1-OP-31.1, IF not done earlier.

4.~~13~~14 At approximately 15% Main Turbine Power:

________ 4.~~13~~14.1 Verify illumination of 1P-E2 "IMP PRESS < 15% AUTO ROD BLK".

________ 4.~~13~~14.2 Verify no control rod motion in process, THEN transfer rod control to MANUAL.

________ 4.~~13~~14.3 Verify that the steam dump control setpoint is at 1005 psig (pot setting of 71.78%), THEN transfer to "Steam Press" control.

________ 4.~~13~~14.4 Transfer the Main Turbine load control to IMP OUT.

NOTE: Closely monitor steam generator levels during this operation.

4.~~14~~15 Verify OR transfer Feedwater control to the Bypass FCV's:

4.~~14~~15.1 Verify the following in AUTO OR under operator control in MANUAL:

________ FCV-1479

________ FCV-1489

________ FCV-1499

4.~~14~~15.2 Place the following in MANUAL AND close:

________ FCV-1478

________ FCV-1488

________ FCV-1498

________ 4.~~15~~16 IF required, transfer auxiliary steam from Unit 1 to Unit 2 main steam OR the auxiliary boilers as per 1-OP-35.1.

Initials

4.0 Procedure (cont.)

4.~~16~~ 17 At 10% Reactor Power:

______ 4.~~16~~ 17.1 Verify illumination of 1P-D3 "AT POWER TRIPS BLOCKED < 10% P-7 PERM".

NOTE: IF the following permissive is NOT lost, note that S.R. Inst. will NOT re-instate AND that Low Power Hi Flux Trip Protection will NOT be available. Refer to Precaution and Limitations 3.5.

______ 4.~~16~~ 17.2 Verify loss of 1P-E1 "PR > 10% P-10 PERM BLK NIS LP TRIPS".

4.~~17~~ 18 IF not previously performed (Step 4.~~10~~ 11) commence removal of the Reheater Steam System by:

______ 4.~~17~~ 18.1 At the reheater control panel,

______ Decrease valve positioner to zero

______ Depress RESET button AND verify closed:

______ FCV-MS-104A

______ FCV-MS-104B

______ FCV-MS-104C

______ FCV-MS-104D

4.~~17~~ 18.2 Align the moisture separator reheaters in preparation of start-up as per 1-OP-28.3.

______ 4.~~18~~ 19 Reduce the Main Generator load to approximately 50 MWe THEN:

______ 4.~~18~~ 19.1 Remove the Main Generator from service as per 1-OP-15.2.

______ 4.~~18~~ 19.2 Verify proper response of the steam dump system.

______ 4.~~19~~ 20 Initiate 1-OP-26.7 to verify OR establish required load shedding scheme while continuing with this procedure.

______ 4.~~20~~ 21 Reduce Reactor Power to ≅ 2%, by manual Control Rod insertion.

______ 4.~~21~~ 22 Note proper response of Steam Dump System.

Initials

4.0 Procedure (cont.)

_______ 4.~~22~~ 23 Stabilize Steam Generator levels at desired level(s).

_______ 4.~~23~~ 24 Remove the Main Turbine from service, as per 1-OP-15.1.

_______ 4.~~24~~ 25 Select, to record on NR-45, the high reading Power AND Intermediate Range Channels.

_______ 4.~~25~~ 26 Proceed to 1-OP-2.1 OR 1-OP-3.1 as required.

Completed By: ______________________

Date: ______________________

PRIMARY-TO-SECONDARY

LEAK RATE SURVEILLANCE (PT-46.2)

- o Check leak rate ---- every 4 hours
- o Check trends ---- every 4 hours
- o Check alarm setpoints ---- every 4 hours

E/13

PRIMARY-TO-SECONDARY

LEAK RATE SURVEILLANCE ITEMS

- N-16 Monitor - every 4 hours
- Condenser Air Ejector Radiation Monitor - every 4 hours
- S/G Blowdown Radiation Monitors - every 4 hours
- Condenser Air Ejector Grab Samples - every 8 hours
- S/G Blowdown isotopic samples (CAP-4.0) - every 24 hours

N-16 MONITOR

- One monitor placed on Main Steam Manifold
- Only monitors high energy gammas (4.5 - 7 MeV)
- N-16 has the following characteristics:
  1. O^{16} (n, p) reaction in core (.05 barns   10 MeV)
  2. 7.10 second half life
  3. 6.1 MeV and 7.1 MeV gammas
- Automatically subtracts background (0.21 cps)
- Conversion factor based on calculations
  1. 99 gpd/cps at 30%
  2. 53 gpd/cps at 50%
  3. 33 gpd/cps at 100%

CONDENSER AIR EJECTOR RADIATION MONITOR

- o Radiation monitor responds to RCS gas activity that leaks through the S/G's and out the condenser air ejector.
- o Correlation between gross count rate and RCS gas activity based on North Anna leak rate data (1983 -1987).
- o Correlation is the same for Unit 1 and Unit 2.

Charlie Willie

Cunningham

S/G BLOWDOWN ISOTOPIC SAMPLES (CAP-4.0)

- Mass balance on RCS and S/G isotopics
    1. I-131
    2. Na-24
    3. Short-lived Iodines
        a. I-132
        b. I-133
        c. I-134
        d. I-135
- Highest value of 1, 2, or 3 above is used.
- CAP-4.0 is currently considered to be the "Keystone" to the leak rate surveillance.

CONDENSER AIR EJECTOR GRAB SAMPLES

- Mass balance performed on RCS gas activity and air ejector gas activity.
- No decay is assumed.
- Mass balance leak rate is performed on:

  1. Xe-135
  2. Kr-87
  3. Ar-41

RADIATION ALARM RESPONSES

- 1-AR-10 (D-) "High" Radiation Alarm annunciator response procedure refers to Standing Order # 155 (if applicable).
- 1-AR-10 (D-4) "High-High" Radiation Alarm annunciator response procedure refers to Standing Order # 155 (if applicable).
- 1-AP-5.1, "Unit 1 Radiation Monitoring System," procedure refers the Operator to immediately notify the STA to evaluate the leak rate trend, and to refer to Standing Order # 155 if the S/G Blowdown monitors, or the Air Ejector Monitor, or the N-16 monitor alarms.

MARTIN MARIETTA

Internal Correspondence

MARTIN MARIETTA ENERGY SYSTEMS, INC.

September 17, 1987

R. W. McClung, 4500S, MS-151

Travel to North Anna Nuclear Power Plant, September 10, 1987

At the request of the NRC, I returned to the North Anna Nuclear Power Plant in Fredericksburg, Virginia. On September 11, I met with Emmett Murphy of the NRC, Ron Ingram of Westinghouse, and Gene Smith and Jim Ogren of the North Anna plant. The trip was made as part of our technical support work under FIN A-9478 "Selected Operating Reactors Issues," Project 3, "On-Call Assistance." The purpose of this trip was to review the status of the eddy-current inspection to data prior to the restart of the unit.

The ruptured tube, R9C51, was pulled and examined by metallography. The cause of failure was determined to be fatigue. The rupture pattern of this tube matched the pattern produced by a tube that was failed in a laboratory fatigue test.

The inspection and data analysis with the 8 × 1 probe was completed on September 10, and the analysis team took a day off before starting work on unit 2. A sample of each type of indication was reviewed: DI (distorted indication at support plate, bobbin coil only), TI (distorted indication at top of tubesheet, bobbin coil only), and PI (possible indication, 8 × 1 probe, anywhere in generator). The calls and their disposition in each case seemed correct to me.

A review of tube R10C51 was also made. This tube was adjacent to the ruptured tube and it was postulated that it might have been damaged by the ruptured tube flopping against it. A possible defect that appeared to be about 20% on the bobbin probe showed no indications on the rotating pancake probe.

No defects or pattern of defects were found that would indicate that there was any mechanism other than fatigue involved in the rupture. Although no defect can be detected in advance, it may be possible to detect a region of cold work in the tube that would give advance warning. The electrical and magnetic properties of metals will change with cold work, and eddy currents have been used to measure this for some metals. A fatigue experiment should be conducted using very accurate equipment to determine the amount of change in the electrical and magnetic properties of Inconel 600 with cold work. Then, if these properties change enough, sufficiently in advance of a rupture, an eddy-current test can be designed to detect fatigue in the tubes. This experiment represents an extended effort and could not be done before the restart.

E/8

To prevent this type of failure in the future, the flow rate in this region of the generator will be reduced and the outer unstabilized tubes will be plugged. The stabilizing bars were found to extend further into the center of the bundle than previously thought. The failed tube was one of the last unstabilized tubes, as determined by an eddy-current location of the stabilizing bars. The eddy-current location of the stabilizing bars with the bobbin probe was a lucky accident rather than by design. The tests are usually designed to ignore things outside the tubing, but in most cases a signal from the 3/8-in.^{2} Inconel bar shows up in the scan. The bar was not present in some of the scans on the outside of the bundle, although it must have been present. The unstabilized tubes toward the outside of the bundle will be preventively plugged. It is likely that some of the tubes to be plugged are actually stabilized but the bar is not being detected. A large rotating pancake probe should detect these bars with no trouble.

The suggestions for detecting the cold work and antivibration bars were passed on to the utility. They may use removable plugs to plug the outer unstabilized tubes and inspect with the large rotating pancake probe at the next outage.

C. V. Dodd

C. V. Dodd, 4500S, MS-151 (4-4839)

CVD:jlb

cc: H. F. Conrad
J. H. DeVan
J. B. Henderson, NRC
A. P. Malinauskas
E. Murphy, NRC ✓
J. G. Pruett
G. M. Slaughter
C. V. Dodd/File

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

The 1S sample selected for the 'C' S/G inspection is based on the following:

- Satisfy 1S 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.

E/89

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

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 - 133	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.

•1980 Refueling Outage

Tubes Inspected	Leakage	Tubes Plugged
S/G A - 485	None	None
S/G B - 476	None	None
S/G C - 478	None	None

Comments: None

•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	None	S/G B - 1
S/G C - 100% available	None	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.

•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 from C S/G. 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.

•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.

-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 and 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.

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%)

September 22, 1987
Type II Letter

Steam Generator Tube Rupture Event
Westinghouse Preliminary Assessment

Summary of North Anna Experience

On July 15, 1987, a steam generator tube rupture event occurred at North Anna Unit 1 shortly after reaching 100% power. For several days prior to the event, air ejector radiation monitor readings were erratic. However, grab samples were taken prior to the tube rupture for environmental release calculations. Subsequent analysis of this data indicated that increasing primary to secondary leakage occurred over a 24 - 36 hour period, which was below normal technical specification limits, prior to the tube rupture event. The ruptured tube was located in Row 9 Column 51 in steam generator "C". The leakage location was found to be at the top tube support plate on the cold leg side of the tube. The opening was circumferential in orientation and extended 360° around the tube. A preliminary assessment indicates that this event does not represent an unreviewed safety issue.

Failure Mechanism Explanation

The cause of the tube rupture has been determined to be high cycle fatigue. The source of the loads is believed to be a combination of a mean stress level in the tube and a superimposed alternating stress. The mean stress is produced by denting of the tube at the top tube support plate and the alternating stress is due to out-of-plane deflection of the tube above the top tube support caused by flow induced vibration. Denting also shifts the maximum tube bending stress to the vicinity of the top tube support plate. These loads are sufficient to produce fatigue and are consistent with a lower bound fatigue curve for the tube material in an AVT water chemistry environment. The magnitude of the alternating stress is consistent with a fluidelastic tube vibration mechanism.

The most significant contributor to the occurrence of excessive vibration is the reduction in damping at the tube-to-tube support plate interface caused by the denting. The absence of antivibration bar (AVB) support is necessary for requisite vibration to occur, together with the reduction in damping. The presence of AVB support will restrict tube motion and thus preclude the deflection amplitude required for fatigue. The original design configuration required AVBs to be inserted to Row 11. Inspection data has shown that some AVBs in the North Anna Unit 1 steam generators penetrate to Row 8, exceeding the minimum AVB insertion depth requirement. No AVB support was present for the Row 9 Column 51 tube that ruptured. Also contributing to the level of vibration, and thus loading, is the local flow field associated with the detailed geometry of the steam generator. The ruptured tube is considered to have a worst case combination of loading conditions and fatigue properties.

The prerequisite conditions derived from the evaluations were concluded to be:

Fatigue Requirements	Prerequisite Conditions
Mean Stress	• Denting

Alternating Stress	• Tube Vibration
	  * Denting at top TSP
	  * High Local Fluid Forces
	  * Absence of AVB support

Material Fatigue Properties	• AVT Environment • Lower range of properties

Steam Generator Evaluation Criteria

A general criteria for operating steam generators has been developed based on the North Anna experience. Any steam generators which have flow conditions and tube support conditions which are less conservative than the modification criteria established for Row 8 through 12 at North Anna Unit 1 should be evaluated.

E/310

The following conditions make up this criteria:

The steam generator should be evaluated if the following conditions (1) & (2) are met
(1) Denting must be present at the top tube support plate,
... and...
(2) the bundle flow parameters must be higher than 90% of those at North Anna prior to modification

This second condition is satisfied if either
(a) the bundle flow is higher than 90% of North Anna,
...or...
(b) the fluidelastic vibration stability ratio is higher than 90% of North Anna.

This criteria is preliminary and believed to be conservative. It is important to note that the North Anna units have the highest bundle flow parameters of the 51 Series steam generators. If a steam generator were found to exceed this criteria, it may still be less severely loaded than the North Anna units prior to modification; yet, further detailed evaluations would be necessary to determine the potential for a similar tube rupture event.

Recommendations
Based on the records available to Westinghouse it appears that your plant falls below the criteria either because (1) denting is not believed to be present at the top tube support plate in any of your steam generators or (2) the bundle flow parameters in your steam generators are less than 90% of the North Anna steam generator values prior to the recent field modification.

Confirm That There is No Denting at the Top Tube Support Plate
As a precaution, it is recommended that the eddy current inspection records for each steam generator be reviewed to confirm that no tube denting is present at the top tube support plate. Denting is believed to be necessary to produce a relatively high mean stress which reduces the tube fatigue endurance limit, to shift the maximum tube bending stress to the vicinity of the top tube support plate and for large amplitude tube vibration to occur.

Actions Recommended for Other Plants
Other plants, with steam generators which exceed the criteria, have been notified to evaluate their plant records describing the condition of all tubes in Row 8 through 12 in all steam generators. Collection of the following data has been recommended:

(1) Identification of all tubes with any denting at the top tube support plate (either hot or cold leg);
(2) Quantification of the AVB insertion depths for each column (from eddy current data);
(3) Notation of any tube wear at any AVB or top tube support plate intersection.

If the results of this data collection confirm that the above criteria are exceeded, then it is also being recommended that the plant systems and practices for determining primary to secondary leak rates be evaluated. Such systems and practices should be capable of producing accurate leak rate data that would detect and classify a tube rupture event like the one that occurred at North Anna.

A Technical Meeting is Planned
The Westinghouse Projects Office will be communicating with you to keep you informed about any further developments relating to your steam generators. A customer meeting to discuss the technical details related to this issue is scheduled for October 16,1987 in Pittsburgh. At that time a detailed presentation will be given describing both the North Anna situation and the potential actions to address and to resolve this issue.

I

. Extensive

EOP - Mult. Tube Ruptures

Procedures were dev. using W guidelines (ERGs) which considered mult. tube rupture.

Operators trained.

Several alarm setpts on air ejector alarm

[illegible] site down

lower -

Rad monitors → shift change

Scheduled (start-up) (Nov 17)

E/11

October 23

Between 4 hrs.

Alarm (Redundant)

Procedure (Abnormal)

~~Start~~ STA - starts ~~[illegible]~~ a surveillance

N-16 on each steam line by first quarter, 88

STA will do the logging

~~blowdown sample~~ - every seven days

grab sample

If SI alarm is received, must recalculate time to exceed 100 gpd.