FINAL RESULTS OF THE

INSERVICE INSPECTION PROGRAM
REFUELING OUTAGE NO. 1

SURRY POWER STATION

APRIL 1, 1975

REPORT NO. ISI 75-3

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Vepco

VIRGINIA ELECTRIC AND POWER COMPANY

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I. INTRODUCTION

In accordance with the requirements of Technical Specification 6.6.C, this report contains a summary of the results of the inservice inspection activities performed during the first refueling outage of Unit No. 1 of the Surry Power Station during the period November 1, 1974 through December 31, 1974.

The document entitled, Inservice Inspection Program, Refueling Outage No. 1, Surry Power Station, Unit No. 1, Report No. ISI 74-1, dated July 18, 1974, provides the specific details concerning the inspections which were scheduled to be performed. Pages 3 through 9 of the aforementioned document summarizes the specific areas to be inspected. Referring to this listing, the following items were omitted or modified during the inspection period:

Component	ISI-261 Ref	Tech Spec Ref	Remarks
Reactor Vessel	1.3	1.3	Will be done by remote UT at a later refueling
	1.4	1.4	Will be done by remote UT at a later refueling
	1.9	1.9	Will be done by remote UT at a later refueling
Associated Aux. Piping	4.9	4.5	A surface examination (PT) was done instead of volumetric (UT) because of configuration

The above deviations comply with the requirements of Technical Specification 4.2.

Some individual welds inspected were different from the ones designated in Report No. ISI 74-1 due to configuration or accessibility. Details of the inspections performed are on file at the Surry Power Station and the Richmond General Office.

The inservice inspections were conducted by Vepco and Westinghouse personnel. The items which the Virginia Electric and Power Company accomplished during the inspection are detailed below:

Component	Tech Spec Ref	Area Inspected	Method of Inspection
Reactor Vessel	1.7	Primary nozzel to safe end weld	VT,PT
	1.8	Closure Studs	PT
Misc. Inspections	7.2	Low head SIS piping in valve pit	VT
	7.3	LP Turbine rotor blades	VT,MT,PT
	8.1.2	Cir. welds and branch connections 4" dia. and smaller	VT

The items which the Westinghouse Electric Corporation accomplished during the inspection are detailed below:

Component	Tech Spec Ref	Area Inspected	Method of Inspection
Reactor Vessel	1.3	Closure Head to Flange	VT/UT
	1.8	Closure studs and nuts	VT/UT
	1.10	Closure washers	VT
	1.13	Closure head cladding	VT/PT
	1.14	Vessel cladding	VT
	1.15	Vessel internals	VT

Component	Tech Spec Ref	Area Inspected	Method of Inspection
Pressurizer	2.1	Circumferential welds	VT/UT
	2.1	Longitudinal welds	VT/UT
	2.6	Manway bolts	VT
		Nozzle to safe end welds	VT/PT/UT
		Skirt weld	VT/PT
Steam Generator Loop A	3.1	Channel Head to Tube Sheet Weld	VT/UT
	3.3	Nozzle to safe end weld	VT/PT/UT
	3.5	Manway bolts	VT
Steam Generator Loop B	3.5	Manway bolts	VT
Steam Generator Loop C	3.3	Nozzle to safe end weld	VT/PT/UT
	3.5	Manway bolts	VT
Reactor Coolant Piping	4.1	Pipe to safe end welds	VT/PT/UT
	4.2	Circumferential Butt Welds	VT/UT
Auxiliary Piping	4.4	Pressure retaining bolting	VT
	4.2	Circumferential Butt Welds	VT/UT
		Nozzle root connections	VT/UT
		Socket Welds	VT/PT
		Nozzle root connections	VT/PT
	4.5	Integrally welded supports	VT/PT
	4.6	Piping supports and hangers	VT
Reactor Coolant Pump (Loops A, B, C)	5.5	Seal Housing Bolting	VT
	5.7	Support structures	VT
	5.8	Flywheels	VT/UT
Valves	6.7	Supports and hangers	VT
Miscellaneous Inspections	8.1.1	Circumferential welds and branch connections larger than 4" diameter	VT/UT
	8.1.3	Socket welds and branch connection welds 4" diameter and smaller	VT/PT
	8.2.1	Containment and recirculation piping	VT/PT
	8.2.2	Remaining sensitized stainless steel piping and cold bends	VT/PT
High Energy Line Piping		Designated high energy line welds in TS Figure 4.15	VT/UT
		All welds exceeding 4" OD TS Figure 4.15 other than designated welds	VT/UT

II. INSPECTION SUMMARY

Inservice examination of Class I components and piping systems, sensitized stainless steel piping, and main steam and feed water piping was accomplished from November 11, 1974 to December 12, 1974. The inspections performed utilized visual, surface, and volumetric nondestructive testing methods. The extent of which systems and components are subject to inspection was established in accordance with ASME Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, with the Summer 1972 addenda. Procedures, UT instruments, UT transducers, UT Calibration Block Certifications, Couplant Certification, Liquid Penetrant Material Certifications, and Personnel Qualifications were reviewed and approved. The arrangement and detail of the Unit No. 1 piping systems and associated components were designed and fabricated before any of the examination requirements of Section XI of the Code were formalized. Consequently, the performance of the examinations has been limited to the extent practical due to accessibility and geometric configuration.

The piping systems of Unit No. 1 contain welds which are inaccessible for examination or examinations are limited to less than 100 per cent of the weld and adjacent base material. Elbow, valve, and tee configuration restricted angle beam examination of the weld and 1T on each side as required by the Code. These welds were generally examined by the following techniques: (1) 100 per cent angle beam of the weld and from the pipe side; (2) longitudinal wave inspection of the pipe side, weld metal, and component areas where search unit contact is possible within the one weld thickness zone; and (3) partial angle beam examination from the component side, search unit contact permitting. This technique satisfies code requirements for inspection of the weld, but does not inspect base metal for 1T on component side of the weld.

The calibration standards utilized for weld inspections were machined from pieces of the same grade material as the systems being inspected. Transfer functions were performed in all material exceeding one inch thickness where required. Transfer corrected gain adjustments were made when required.

The auxiliary piping examination results record notes examination restrictions due to geometry and obstructions. Removal of existing hangers for access was not considered because of the high risk of stressing adjacent piping weldments in the system.

A volumetric examination of integrally welded pipe supports was not accomplished since meaningful volumetric examination of these supports cannot be made with present techniques. As a substitute, a liquid penetrant and a visual examination was performed on those supports which would have been examined by ultrasonics.

Liquid penetrant examination of the sensitized piping cold bends was accomplished by examining a minimum area of one inch wide by one foot long on each bend and recording the location of the area examined.

Ultrasonic examination of the main steam designated pipe welds, Loop 1 Weld Number 1, Loop 2 Weld Number 155, Loop 3 Weld Number 275, could not be accomplished as these welds are well inside the containment wall penetrations and are not accessible. As a substitute for these welds, the next available weld in the main steam valve house was ultrasonically examined. These welds are designated Westinghouse Weld Number 13 for Loops 1, 2 and 3.

III. INSPECTION RESULTS

The inservice inspections performed during the period covered by this report included the basic areas listed below:

1. Reactor coolant system, including reactor vessel, pressurizer, steam generator welding and bolting, auxiliary piping, reactor coolant piping and valves.
2. Piping systems containing sensitized stainless steel, including safety injection system, charging system, reactor coolant system (lines less than 4 inches in diameter), containment spray system, recirculation spray system, and other miscellaneous piping containing sensitized stainless steel.
3. Designated high energy line welds described in Technical Specification 4.15.
4. Low pressure turbine rotor blades.
5. Materials irradiation surveillance capsule.
6. Steam generator tube inspections.
7. Reactor Internals Inspection.

Results of each of the above inspections are summarized below.

1. Reactor Coolant System

Inservice examination of components and piping systems within this area were performed by the Westinghouse Electric Corporation. The inspections performed utilized visual, surface and volumetric non-destructive testing methods. The results of examinations performed by the three methods and disposition of any indication are listed below.

a. Volumetric Examinations

Volumetric examinations performed did not reveal any rejectable flaw indications.

b. Surface Examinations

Surface examinations revealed the following linear indications:

SYSTEM	ISOMETRIC DRAWING NO.	WELD NO.
Loop 1 - 2" Fill Header	RC-198-1502	2
Loop 2 - 2" Drain Header	RC-57-1502	3
Loop 3 - 2" Fill Header	RC-200-1502	2

All indications were located in the base material of the adjacent casting. The indications were ground out, repaired and re-inspected by liquid penetrant. The re-inspections showed no rejectable indications.

c. Visual Examination

Visual examination performed revealed the following conditions:

SYSTEM	ITEM	CONDITIONS
Loop 3 - Drain Header	Welded support	Arc Strike
2" Drain Header	All welded channels and U-bolts	Carbon steel
	Support bracket T	Broken plate
Loop 3 - Seal Injection	Pipe strap rod A	Broken rod
Loop 3 - Pressurizer Spray	Spring hangers R	Readings off scale maximum
Loop 3 - Accumulator Discharge	Spring hangers D	Readings off scale maximum

SYSTEM	ITEM	CONDITIONS
Loop 3 - Accumulator Discharge	Spring hangers C	Readings off scale maximum
Loop 3 - Pressurizer Surge	Spring hangers C	Readings off scale maximum

The indications on the welded support on the Loop 3 drain header were removed and re-inspected by liquid penetrant and visual examination. Re-inspection showed no indications. The remaining items were repaired and/or adjusted to proper setting.

Vepco personnel performed the examination of the reactor vessel closure studs and the primary nozzles to safe end welds by visual and/or liquid penetrant non-destructive test methods. The results of the examinations were acceptable, with only small rounded indications on the primary nozzles to safe end welds indicated by liquid penetrant examinations.

2. Piping Systems Containing Sensitized Stainless Steel

Piping systems containing sensitized stainless steel were visually examined. A number of arc strikes were noted in the valve pit area. The indicated arc strikes are not significant; however, they will be removed for future inspections.

3. High Energy Line Piping

The high energy line welds designated in Technical Specification 4.15 were volumetrically examined by ultrasonic non-destructive testing methods. No discrepancies were noted.

4. Low Pressure Turbine Rotor Blading

The low pressure turbine blading was examined by visual and surface non-destructive testing methods. The results of the inspection and disposition are summarized below:

AREAS EXAMINED	METHOD OF EXAMINATION	INDICATION	DISPOSITION
Blading	Visual	Arc Strikes	Ground out and re-inspected satisfactorily
	Magnetic particle	5 cracked last stage blades	Blades were replaced
Stellite erosion shields	Visual	One shield missing	Replaced shield
Lashing lugs	Liquid Penetrant	Cracked lashing lugs	Ground out and repair welded. Re-inspected satisfactorily.
Undershroud welds	Liquid Penetrant	Cracking in undershroud welds	Ground out and repair welded. Re-inspected satisfactorily.

5. Material Irradiation Surveillance Capsule

Battelle Columbus Laboratories is presently examining the first surveillance capsule. The Technical Specification requirements were satisfied.

6. Steam Generator Tube Examinations

During the refueling outage, Eddy current inspections were performed on all three steam generators. In conjunction with the conversion to all volatile treatment of the feedwater, sludge lacing was accomplished in parallel with the inspection program.

The inspection program required examinations at 400 KHZ to detect and measure potential tube defects with supplemental examinations at

100 KHZ to provide an assessment of low level wall thinning and at 25 KHZ to measure sludge deposits on top of the tubesheet. A nominal 100 per cent inspection was also performed on the inlet side of steam generator A up to the first support. This was accomplished along with U-bend inspections of peripheral tubes. Based on the results of the planned inspection on steam generator A, the inspection program for steam generators B and C was expanded to include those areas of the tubesheet array where defects were noted in steam generator A.

A total of 3852 tube inspections, including inlet and outlet, were performed at 400 KHZ in steam generator A. Additionally, 1243 tube inspections were performed in steam generator C. As a result of this inspection program, 55 tubes were explosively plugged in steam generator A, 2 in B and 84 tubes in C steam generator. All those tubes plugged in steam generator A exhibited defects of 50 per cent or greater. The two tubes plugged in B steam generator having defects of 30 per cent and 38 per cent were plugged. Two of the 84 tubes plugged in steam generator C had defects of 25 per cent and 47 per cent. All others exhibited defects of 50 per cent or greater.

A summary of the inspection results for each steam generator is given below:

Steam Generator A

A total of 3262 tubes were examined at 400 KHZ from the inlet side. Of these tubes, 271 exhibited detectable wall penetration (i.e. >20 per cent), 55 defects were equal to or greater than 50 per cent, 81 were between 40 and 49 per cent, 92 were between 30 and 39 per cent and 43 fell in the range of 21 to 29 per cent. An additional 80 tubes were noted to have probable defects less than or equal to 20 per cent.

Outlet inspections of 400 KHZ were performed in 580 tubes. Only 4 tubes were noted to have defects in excess of 20 per cent and these were confined to the 21 and 29 per cent range. Three others exhibited defects of 20 per cent or less.

Crud level measurements were made from both inlet and outlet sides at 25 KHZ prior to sludge lancing. Following the lancing, 25 KHZ sludge measurements were performed from the inlet side only.

Steam Generator B

Inlet side inspections were made at 400 KHZ to a total of 1204 tubes. Three of these tubes exhibited detectable wall penetration, two were in the range of 21 to 29 per cent and one between 30 and 39 per cent. Another 39 tubes were noted to have defects less than or equal to 20 per cent. No unacceptable defects were noted.

Thirty-nine tubes were inspected from the outlet side with no defects noted.

Crud level measurements were made from both inlet and outlet sides following the sludge lancing effort.

Steam Generator C

Of the 1514 tube inspections performed at 400 KHZ from the inlet side, 206 tubes exhibited detectable wall penetration (i.e. >20 per cent). Eighty-two tubes had defects of 50 per cent or greater, 66 tube defects fell in the range of 40 to 49 per cent, 39 defects in the range of 30 to 39 per cent and 19 tubes had defects in the range of 21 to 29 per cent. Another 57 tubes were noted to have defects less than or equal to 20 per cent.

A total of 3 detectable defects were noted in the 84 tubes examined from the outlet side. All of these were in the 21 to 29 per cent range. Another 4 defects were detected and assessed to be less than or equal to 20 per cent.

Crud level measurements were made from both inlet and outlet sides following the sludge lancing program.

7. <u>Reactor Internals Inspection</u>

During the refueling all of the fuel assemblies were removed from the reactor vessel. As a result, the opportunity was taken to perform an inspection of the reactor internals and core components. However, the reactor lower internals were not removed; therefore, the outside of the lower internals and certain other parts of the reactor vessel were not inspected. The Technical Specifications requirements were satisfied. The emphasis of the inspection was on the upper internals, accessible areas of the lower internals, vessel clad, and selected control rod drive shafts and rod cluster control assemblies (RCCA's).

All inspections were carried out by certified inspectors using closed circuit underwater television systems and recorded on video tape. The reactor internals were found to be in good condition and only one defect was noted. The locking cups for fasteners of the upper guide tube on the upper internals at location D-4 had not been crimped during construction. These locking cups were crimped in the field utilizing approved procedures.

Information of particular importance in the evaluation of each of the reactor components inspected included the following:

a. Condition of wear and alignment surfaces.

b. Integrity of critical structural joints.

c. Condition of mechanical fasteners, alignment pins and locking devices.

d. Free movement of movable components.

e. Corrosion product formations.

f. Presence of debris.

g. Mechanical distortion

Components included in the evaluation were the following:

a. Reactor upper internals

b. Reactor lower internals

c. Drive line components

d. Reactor vessel

The program included the inspection and evaluation of certain important areas of the internals which are good indicators of performance. The overall effects of conditions found were evaluated on the basis of design information, experimental results and previous experience with other reactors.

A small diameter television camera was lowered through the irradiation specimen access holes at the 65 degree and 285 degree locations for the purpose of inspecting the vessel clad at these locations. Two six (6) by six (6) inch patches were inspected at each location. One patch at each location was adjacent to and above the top of the irradiation specimen basket at each location. The second patch was approximately six (6) feet higher. All four patches of vessel clad appeared to be in excellent condition.

IV. CONCLUSIONS

The results of the inservice inspections performed verified the integrity of the systems and components examined and satisfied the requirements of Technical Specification 4.2. The discrepancies noted were corrected.

Based on the results of the inservice inspection program, as summarized herein, the safety systems and components inspected have not experienced degradation and there is reasonable assurance that they will continue to perform their design function in a safe and satisfactory manner.

V. REFERENCES

1. Reactor Internals and Core Components Evaluation, Surry Unit 1, November 1974, Refueling Shutdown, March 1975, Westinghouse Electric Corporation.

2. Report of Inservice Inspection Conducted by Vepco Personnel, Refueling Outage No. 1, Unit No. 1, Surry Power Station, March 1, 1975, Virginia Electric and Power Company.

3. Inservice Inspection Report of the Surry Unit No. 1 Nuclear Power Station, January 1975, Westinghouse Electric Corporation.

4. Inservice Inspection Program, Refueling Outage No. 1, Surry Power Station, Unit No. 1, ISI 74-1, July 18, 1974, Virginia Electric and Power Company.

5. Section 4.2, Technical Specifications, Surry Power Station, Unit Nos. 1 and 2, Virginia Electric and Power Company.

6. Field Service Report MRS 4.4 (VPA-2), Surry No. 1-VPA, Westinghouse Electric Corporation.

7. Battelle Columbus Laboratories letter of April 22, 1975.