Review of Section XI Inservice Inspection Program Effectiveness

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ABSTRACT

To evaluate the effectiveness of Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, searches were performed of the Licensing Event Report and Nuclear Plant Reliability Data System computerized data bases, and a review was made of inservice inspection summary reports. It was found that the Section XI examinations and tests detect flaws in welds and plant components and result in subsequent corrective action. This study also shows that the format and topics of information provided in Section XI-prescribed inservice inspection summary reports vary widely.

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SUMMARY

This is the fifth in a series of reports evaluating inservice inspection (ISI) examinations of commercial nuclear power plants. This work was performed by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory (INEL) for the Nuclear Regulatory Commission (NRC) under Task 24 of FIN A6367. The data studied for this report were compiled from licensee inservice inspection summary reports, the Licensing Event Report (LER) data base and the Nuclear Plant Reliability Data System (NPRDS) data base.

The purpose of this phase of Task 24 was to further evaluate the effectiveness of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code examination requirements and to recommend changes for improvement if necessary.

This study revealed that ASME B&PV Section XI Code examinations are detecting relevant conditions in welds and plant components. This study also identified the need to clarify some requirements in certain subsections of Section XI.

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Review of Section XI Inservice Inspection Program Effectiveness

INTRODUCTION

This is the fifth in a series of reports on the effectiveness of inservice inspection (ISI) examinations performed by commercial nuclear power plants per the requirements of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code.¹ The first report² evaluated examinations performed by 12 nuclear facilities and reviewed 96 Licensee Event Report (LER) abstracts to determine if Section XI requirements were instrumental in discovering conditions reported. The second report³ was a review of the ISI administrative system, the third report⁴ evaluated the effectiveness of Section XI steam generator examination requirements, and the fourth report⁵ evaluated additional ISI summary reports to determine the effectiveness of Section XI examinations and develop recommendations for ASME Code revisions. The purpose of this phase of Task 24 of FIN A6367 was to further evaluate ISI effectiveness by reviewing operational data in connection with Licensing Event Reports (LERs), the Nuclear Plant Reliability Data System (NPRDS), and additional ISI summary reports to determine the effectiveness of Section XI examinations and develop recommendations, if necessary, for Section XI ASME B&PV Code revisions.

This report includes a compilation and analysis of data contained in LERs, the NPRDS, and selected summary reports of ISI examinations that licensees submit to the NRC. These reports are due within 90 days of the end of each series of examinations performed during a refueling outage.

DATA BASE SEARCH TECHNIQUES AND RESULTS

The LER data base was searched for records that included detection by inspection/testing and 'Section XI' in the abstract. It was necessary to perform a text scan of the abstracts since no field in the LER data base covers Section XI directly. Seventy of 38,097 LERs met this search criterion. Most of these dealt with Section XI inservice testing of pumps and valves or referenced Section XI for repair and replacement. Since this report deals with the effectiveness of Section XI nondestructive examinations, the testing and repair/replacement LERs were not reviewed extensively. The 17 LERs that reference Section XI examinations are included in Appendix A and summarized in Table 1. In addition, Appendix A also includes four additional LERs (Form Numbers 48, 50, 55, and 58) that reference Section XI, but do not pertain to examinations.

The search conditions used for the NPRDS was safety class 1, 2, or 3 and detection by inservice inspection or testing. This provided 3,038 failures from a total of 146,107 failure records. Details for 99 pipe failures and 25 vessel failures in this group were then printed to disk, edited to clean up the format, and further analyzed. These records are provided in Appendix B. Table 2 summarizes the NDE methods referenced in the records contained in Appendix B.

An LER record may contain multiple components, but each component will have a single record in the NPRDS system. More data were found in the NPRDS search, since having "Section XI" in the text was not a limitation used in the NPRDS search. Some data in the NPRDS match records in the LER data base (e.g., the Form 7 LER corresponds with the last pipe record for Millstone 1 in Appendix B). However, the Form 8 LER did not show up in the NPRDS search.

These data bases provide a much quicker means of locating the information on Section XI examinations that find significant conditions. This is because only one plant and about one year of examinations are included in any single summary report and because the summary reports cannot be searched using a computer, while the data bases cover all plants for the history of the plants and are searched using a computer.

Table 1. Summary of LER Information

LER Form No.	Component	Condition Found	NDE Method*
3	main steam piping welds	fabrication flaws	RT
4	feedwater system piping welds	fabrication flaws	RT
5	pipe supports	failed attachment welds	VT
6	steam generator tubing	stress corrosion cracking and pitting	ET
7	class 1 stainless piping	crack in weld	NS
8	piping	inclusions	UT
11	valve	weld-related flaw	UT
16	nozzle to shell weld	slag in weld root area	UT
30	piping	improper valve lineup	VT
32	piping	intergranular stress corrosion cracking	UT
44	RHR class 1 piping	flaw (leak) in plug type repair weld	VT
45	piping welds	fabrication flaws	RT
52	class 1 piping welds	unacceptable flaw	NS
57	steam generator tubing	stress corrosion cracking and pitting	ET
61	check valves	plugging	VT
62	check valve	missing spring	VT
69	steam generator tubing	stress corrosion cracking and pitting	ET

* ET, RT, UT, VT and NS respectively indicate eddy current, radiographic, ultrasonic, visual and not specified nondestructive examination methods.

Table 2. Distribution of NPRDS Referenced NDE Methods*

Component	LT	RT	PT	VT	UT	NS
class 1 piping			18	7	25	8
class 2 piping	1	2	11	10	9	2
class 3 piping	2			4	1	
vessels			1	23	1	
Total	3	2	30	44	36	10

* LT, RT, PT, UT, VT and NS respectively indicate leak or flow test, radiography, penetrant, ultrasonic, visual and not specified nondestructive examination methods.

ISI SUMMARY REPORTS

Information was compiled from 15 ISI summary reports of 15 nuclear facilities (see Appendix D for excerpts from selected summary reports). Twelve of these facilities were pressurized water reactors (PWRs) and three were boiling water reactors (BWRs). Four reactor suppliers are represented, (i.e., six were Westinghouse (W) and three each were General Electric (GE), Combustion Engineering (CE), and Babcock and Wilcox (BW). Table 1 shows the different edition/addenda of Section XI of the ASME B&PV Code referenced in the reviewed ISI summary reports.

Table 3. Distribution of Section XI Code Edition/Addenda.

80-W80*	80-W81	83-S83	86 edition
2 plants	5 plants	6 plants	2 plants

* Indicates the 1980 Edition with the Winter 1980 Addenda.

Review comments on the 15 selected summary reports follow:

Plant A (B&W) - 80-W80 Code. This includes the NIS-1 form with 398 pages following that are not uniformly numbered. No NIS-2's are contained as they are not required by the Code addenda used. A computerized seven-page repair/replacement log is supplied instead. 'Alternative' inspections are included in a status table but no definition of 'alternative' is supplied. These apparently are examinations performed in lieu of the Code-required examinations and covered by relief requests. A 103-page inspection plan is included, followed by 56 pages of computerized results, with relief requests referenced for limited coverage examinations. This is information that should be a regulation requirement for all summary reports. Narrative Class 1 and 2, plus augmented and alternative examination results, then follow and are shown in Appendix C. Much ultrasonic data on resolving indications in components using amplitude drop sizing techniques that have long been known to have limitations, but are specified by the Code addenda used, are included. Another problem resolution, including UT sizing using a tip diffraction technique, is included. A fracture mechanics analysis report for resolution of one UT detected flaw is included, but it is marked "Proprietary."

Plant B (GE) - 80-W80 Code. Class 3 results are included although not required by the Code. A table of ISI inspection periods, and outage history and plans for the interval are included. This would be good to have as a Code requirement for inclusion in all summary reports. Augmented (noncode) examinations are included. GE did many of the RPV exams under contract. The GERIS automated system was used on nozzle-to-vessel welds. A summary table of examination limitations is provided with a commitment to either complete the required examinations at a later outage or request relief. A table of 'significant' indications is included. An extensive package of NIS-2 forms take most of the report volume, but a computer-generated summary table of repairs and replacements is included that provides the information in a much

more usable format than the NIS-2 forms. This is a better format for the information in a summary report, and a Code change to eliminate the NIS-2 form may be beneficial.

Plant C (B&W) - 80-W81 Code. This report was about a month later than the Code required 90 days after examination completion. It does reference a letter that requested an extension due to an unexpected heavy workload. It includes an examination table with no result or limitations included (another 'plan-type' table similar to Indian Point 3). Qualitative results are included in the text on supports (a few with minor problems), augmented examinations for IGSCC, pure water stress corrosion cracking, and other piping weld indications (dripping associated with crack-like indications on noncode, nonsafety piping). Apparent slag indications reexamined in reactor coolant pumps were reported as 'shrinking in size due to refinements made in the UT equipment.' It discusses use of Code Case N-481, which provides for analysis in lieu of examination of reactor coolant pump casing welds.

Plant D (W) - 80-W81 Code. This includes a technical report justifying an IWA-2240 alternative ultrasonic examination calibration block. This is a case where an alternative to a specific Code requirement is permitted by the Code and a relief request to NRC is not necessary. A summary of repairs and replacements is provided.

Plant E (CE) - 80-W81 Code. The NIS-1 form references a summary report that includes items 1 through 6 from the NIS-1 form on every page. Class 3 results are included, even though they are not required by the Code. A summary table (index) of the NIS-2 forms is included. Review of the section, "Abstract of Conditions Noted and Corrective Actions Taken," showed that rejectable piping welds were found by surface examinations and that the indications were removed by grinding. The examination sample was not expanded as required by the Code. They state: "The initial indications on both welds were most likely left over from construction. No additional welds were inspected as a result of these minor indications."

Plant F (GE) - 80-W81 Code. The cover letter summarizes several augmented examinations. The report takes an exception to the NIS-1 requirement that supplemental sheets include 6 items of information. Other reports also do not follow that requirement. A change to the form should be initiated to resolve this issue. Many contractors were used. EPRI-qualified UT examiners were used for all examination of IGSCC-susceptible materials. Clad overlay repairs are discussed. A summary table of all reportable indications is included. The examination summary tables do include results and quantitative limitations. The report does not include Form NIS-2, since it is not required by the Code used, but it includes a summary table of repairs and replacements. This is a better format for the information in a summary report, and a Code change to eliminate the NIS-2 form may be beneficial.

Plant G (GE) - 80-W81 Code. A useful summary of examinations and results broken down into broad categories is included in front of the NIS-1 form, which contained much of the same information repeated in a less usable form. A narrative of the repairs and replacements, plus a summary, is provided which facilitates review. Indications are noted, but there is no reference to expanded examination samples as required by Section XI of the ASME B&PV Code. Relief requests are referenced on the examination data tables included with the NIS-1 form.

Plant H (W) - 83-S83 Code. Results are included in a notes section, which follows the examination summary table. A brief index to the NIS-2 forms is provided. A note is made regarding a 25% through-wall limitation of cast stainless steel ultrasonic examinations with a relief request referenced.

Plant I (W) - 83-S83 Code. Another problem similar to the Plant E regarding not doing required sample expansion was found. Results are included in a notes section that follows the examination summary table. A brief index to the NIS-2 forms is provided. Reference to other NRC submittals on steam generator tubing examinations, fracture mechanics analysis of flaws in nozzle-to-vessel welds, and relief requests is made. A note is made regarding a 25% through-wall limitation of cast stainless steel ultrasonic examinations with a relief request referenced.

Plant J (CE) - 83-S83 Code. Another large package, 63 NIS-2s, were not included due to 'administrative and clerical' problems. They committed to supply the missing NIS-2s with the next ISI summary report. Page 4 of the NIS-1 (Appendix C) gives a brief summary of the ISI results. SWRI was ISI contractor. The organization supplied a three-volume report that was submitted in total. Useful examination summary tables with results are provided. Additional unnecessary detail for a summary report includes handwritten individual examination data sheets and copies of examination personnel certifications.

Plant K (CE) - 83-S83 Code. No index of the 1 in. thick stack of NIS-2s was provided. A component-by-component summary of results in the format supplied by the contractor, SWRI, is provided. This requires going through an extensive page-by-page and line-by-line search to find where ISI detected problems. A brief half-page summary of steam generator eddy current tests is provided.

Plant L (W) - 83-S83 Code. Dynacon Systems was contractor. This summary report consisted of extracts from the Dynacon-supplied ISI report. The 58 pages included about 33 pages of copies of the Code examination requirement figures. It included tables of planned examinations, without results or examination limitations included. It includes the required NIS-1 Form, but no NIS-2 forms as required by Section XI. It references the 1986 edition of Section XI for use of Appendix VI, which covers performance demonstration for UT of bolting. It briefly mentions 36 recordable indications, 10 surface (components not given) and 26 visual (all on IWF supports). It mentions limited examinations (geometry, access, etc.), but does not give details or reference relief requests.

Plant M (W) - 83-S83 Code. SWRI was contractor. Portions of the SWRI report to the licensee are included; some appendices that include details not required in summary reports were missing. The completed form NIS-1 includes normal ISI, steam generator ET results, and IST results. Usually, these are in separate reports and submittals. A summary table of ISI results is included. Examinations not required by Section XI for erosion/corrosion are discussed. One indication was found outside the Code examination area, but on a Code component. The disposition (Appendix C) states that the material removal specified to remove the indication does not require a Code job package. Since this is a Code item, a Section XI Code repair should have been performed. A summary of the repairs and replacements is provided. Some are covered by NIS-2 forms, but most are covered by NR-1 forms, which reference an NR stamp and are available from the National Board.

Plant N (W) - 86 Code. This was the largest volume summary report reviewed (an 11 1/2 in. thick pile of paper, including a 2 in. thick steam generator eddy current report prepared by CE). It includes the ET procedure, the employer-written practice for personnel certification, plus all of the personnel certifications, which are proprietary. The licensee submitted the entire contractor report rather than providing a summary report. In a similar manner all the ISI results, three volumes, prepared by the contractor NES, make up the remaining parts of the report. There are about 117 pages of NIS-2 forms with no summary table. Excessive detail such as the examination plan, individual result sheets for each examination, and calibration data sheets are included. This is not a summary report, but the original vendor-supplied plant ISI data. One item of interest discovered is an evaluation of a nozzle indication, which involved extensive resources with multiple UT examinations by different organizations plus radiography to find whether the indication was an inservice flaw or an existing fabrication flaw (see Appendix C).

Plant O (B&W) - 86 Code. This report covers both first- and second-interval examinations that occurred during overlapping dates in the same refueling outage. It appears that this practice may violate the intent of the Code. A plant should be operated for a minimum of one refueling outage before credit can be taken for examinations performed in the interval. The Code requirements need to be expanded to clarify the intent. The report lists at least one examination of a reactor vessel shell-to-flange weld that is credited for both intervals when only one examination was performed. There are other duplicate credits of nozzle-to-shell welds and nozzle radii. All duplicates could be found if a computer search were available. It would be beneficial to the licensees and NRC if the summary information were submitted in computerized form. The examination summary tables do include information on examination limitations, and a summary table of indications, showing all but one indication associated with supports, is provided. Only six pages of NIS-2 forms are attached.

The above review comments and the excerpted summary reports in Appendix C show wide interpretation of what should be in the ISI summary report and show it may be beneficial to not require submittal of the NIS-2 forms, but to use a summary table of the repairs and replacements for the summary report. Also, the large volume of data could be better submitted on a computerized format, which would both reduce paper work and facilitate review. Future effort should be applied to prepare a Section XI appendix on format and content of summary reports. This action should also include an option for submitting computerized data.

The first Section XI of the ASME B&PV Code published in 1971 was written for Class 1 and required that summary reports for all examinations and tests be submitted to the NRC. The Winter 1972 Addenda added Class 2 and only required reports for examinations, not tests. Further, it only required reports for examinations that resulted in repairs or replacements. The Summer 1973 Addenda incorporated Subsections IWP (pumps) and IWV (valves). Class 3 was added in the 1974 Edition. Article IWA-6000, Records and Reports, of the 1974 Edition required summary reports to be submitted for all examinations and tests for Class 1 and Class 2 components, but excluded Class 3. It is possible that the addition of Class 3 was overlooked by the group responsible for IWA-6000. Correction of this omission is in progress in the ASME B&PV Code Committee.

Subsubarticle IWA-6210, Owner's Responsibility, presently requires that Owners prepare preservice and inservice inspection summary reports for all Class 1 and Class 2 pressure-retaining

components and their supports. Subsubarticle IWA-6220, Preparation, states that inservice inspection summary reports shall be prepared at the completion of each inspection conducted during a refueling outage and that examinations, tests, replacements, and repairs conducted since the preceding summary report shall be included. Subsubarticle IWA-6230, Summary Report Submittal, requires the Owner to file inservice inspection summary reports with the enforcement and regulatory authorities having jurisdiction at the plant site within 90 days of completion of the inservice inspection conducted during each refueling outage.

The Plant C licensee missed the IWA-6230 deadline for summary report submittal by about a month, but referenced a letter that requested an extension due to an unexpected heavy workload.

The Plant O licensee performed examinations for the first and second intervals in one refueling outage and provided a joint summary report for the two intervals. This joint submittal included credit for one examination counting for two intervals. This practice does not comply with the intent of Section XI, and B&PV Code action to resolve this issue has been initiated as shown in Appendix E. Some licensees are misinterpreting the provisions of IWA-2430(d) of Section XI, which permits extending or decreasing an inspection interval by as much as one year to provide overlapping of intervals.

Subsubarticle IWA-6220, which requires that summary reports include all examinations and tests for Class 1 and Class 2 components, is being misinterpreted by many facilities. None of the facilities in this study included results of testing pumps or valves. Resolution of this problem will now requires action by the ASME Committee on Operation and Maintenance (OM) of Nuclear Power Plants, since inservice testing of pumps and valves is now controlled by the ASME OM Code. The ASME OM Code does not presently require summary reports.

Though required to report only on Class 1 and Class 2 components, the Plant B and Plant E summary reports included Class 3 results. Previously, 11 of 27 reports reviewed contained Class 3 results. Section XI should require the summary reports to include the results of all examinations and tests, regardless of class. The proposed change to accomplish this is in progress in the B&PV Committee. This is a low-impact change, since IWA-6210(b) already requires preparation of the records used to make up the summary report.

The Plant E and Plant I licensees did not expand the examination sample as required by Section XI when rejectable results are found in the initial sample.

Some of reports included augmented examinations, although this is not a requirement. Inclusion of augmented examinations in all summary reports should be required in the proposed FIN A6367 Task 38 action to place the Section XI submittal requirements to the NRC in 10 CFR 50.55a.

Below are some NRC documents that licensees reference for augmented examinations:

1. NRC Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of Power Plants"
2. NRC I.E. Bulletin 79-13, "Cracking in Feedwater System Piping"

3. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"
4. NRC Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants"
5. Summary NUREG 0578, "TMI Lessons Learned"
6. NRC Bulletin 82-03, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter Stainless Steel, Recirculation System Piping at BWR Plants"
7. NRC Bulletin 83-02, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter Stainless Steel, Recirculation System Piping at BWR Plants"

A decision that resulted from this study is that Title 10 Part 50 of the Code of Federal regulations should be amended. 10 CFR § 50.55a (g)(5)(IV) states that where an examination or test is determined to be impractical and not included in the licensee's revised ISI program, the relief requests shall be submitted not later than 12 months after the 120-month interval. A facility can repeatedly not perform a required examination for 11 years. Also, by the time a problem is identified, after the completion of the ISI interval, it is too late to implement corrective action.

It is suggested that 10 CFR § 50.55a be revised to require that relief requests either be referenced in or be submitted with the summary reports.

CONCLUSIONS

This study revealed that Section XI examinations are of value in detecting relevant conditions in welds and plant components.

Problems were identified regarding ISI interval scheduling of examinations and examination sample expansion. Also, Section XI should be changed to include all Section XI examination and test results in the summary reports.

Confusion regarding the content of ISI summary reports was noted. Future effort should be applied to prepare a Section XI appendix on format and content of summary reports. This action should also include an option for submittal of computerized data to reduce paper volume and facilitate information review.

It may be beneficial to not require submittal of the NIS-2 forms, but to use a summary table of the repairs and replacements for the summary report.

Another recommendation, because limited examinations were reported that were not covered by relief requests, is to change 10 CFR § 50.55a to require relief requests to be referenced or submitted with the summary reports so that corrective action or substitute examinations can be identified and implemented before the start of the next inspection interval.

REFERENCES

1. "American Society of Mechanical Engineers Boiler and Pressure Vessel Code," Section XI, Division 1.
2. D. A. Aldrich, J. F. Cook, *Evaluation of ASME Code, Section XI, Division 1, Inservice Inspection*, EGG-SD-7797, December 1987.
3. J. F. Cook, *Review of the Inservice Inspection Administrative System*, EGG-MS-8130, August 1988.
4. D. A. Aldrich, J. F. Cook, *Effectiveness of Steam Generator Inservice Examinations*, EGG-MS-8417, April 1989.
5. D. A. Aldrich, J. F. Cook, *Effectiveness of Inservice Inspection Examinations*, EGG-MS-8848, May 1990.

Appendix A

LER Information

Appendix A

LER Information

FORM 3 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
206 1980 024 1 8006030434 166743 05/16/80

DOCKET:206 SAN ONOFRE 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SOUTHERN CALIFORNIA EDISON CO.
SYMBOL: SCE

ABSTRACT

POWER LEVEL - 000%. CAUSE - INSTALLATION ERROR. WHILE PERFORMING RADIOGRAPHIC EXAMINATION OF MAIN STEAM LINES INSIDE CONTAINMENT, SIX WELDS WERE FOUND WITH INDICATIONS IN EXCESS OF THOSE PERMITTED BY ASME B&PV CODE, SECTION I. FIVE OF THE WELDS HAVE FLAWS CONSISTING OF SLAG INCLUSIONS AND/OR LACK OF PENETRATION. ONE WELD HAS THREE SMALL CRACKS IN THE ROOT. THE INDICATIONS APPARENTLY HAVE EXISTED IN THE LINE FROM ORIGINAL PLANT CONSTRUCTION. ALL WELDS WERE REPAIRED UNDER SECTION XI REQUIREMENTS.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PC --CONSTRUCTION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:FA --MAIN STEAM
SECONDARY SYSTEM:SA --PRIMARY REACTOR CONTAINMENT (PWR)
COMPONENT:PIEC--PIPE, GREATER THAN 16 INCH DIAMETER, CARBON STEEL

VENDOR:B130--BECHTEL CORP.
QUANTITY:5 --
TRAIN:Z --UNKNOWN
TIMING:P --POTENTIAL
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BP --EXTERNAL SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:FA --MAIN STEAM

COMPONENT:PIEC--PIPE, GREATER THAN 16 INCH DIAMETER, CARBON STEEL

VENDOR:B130--BECHTEL CORP.
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 4 LER SCSS DATA 12-24-92

DOCKET	YEAR	LER	NUMBER	REVISION	DCS	NUMBER	NSIC	EVENT	DATE
206	1980	027	0	8007010537	160255	05/18/80			

DOCKET:206 SAN ONOFRE 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: SOUTHERN CALIFORNIA EDISON CO.
SYMBOL: SCE

ABSTRACT

POWER LEVEL - 000%. CAUSE - FAULTY WELDS. RADIOGRAPHIC EXAMINATION OF

FEEDWATER SYSTEM WELDS INSIDE CONTAINMENT FOUND SEVEN (7)

UNACCEPTABLE DEFECTS. FIVE (5) CASES OF LACK OF PENETRATION, ONE (1)

CASE OF POROSITY, AND ONE (1) CASE OF SLAG INCLUSION. ALL DEFECTS WERE ORIGINAL CONSTRUCTION DEFECTS. SEVEN (7) WELDS WERE REPAIRED UNDER THE REPAIR REQUIREMENTS OF ASME B&PV CODE SECTION XI.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PC --CONSTRUCTION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:FI --CONDENSATE AND FEEDWATER
SECONDARY SYSTEM:SA --PRIMARY REACTOR CONTAINMENT (PWR)
COMPONENT:PICZ--PIPE, >6 AND <= 10 INCH DIAMETER, UNKNOWN MATERI
VENDOR:B130--BECHTEL CORP.
QUANTITY:7 --
TRAIN:Z --UNKNOWN
TIMING:P --POTENTIAL
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BP --EXTERNAL SYSTEM LEAKAGE, UNKNOWN RATE

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 5 LER SCSS DATA 12-24-92

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
206	1982	009	0	8204220328 173061		04/05/82

DOCKET:206 SAN ONOFRE 1 TYPE:PWR
REGION: 5 NSSS:WE
ARCHITECTURAL ENGINEER: BECH

FACILITY OPERATOR: SOUTHERN CALIFORNIA EDISON CO.
SYMBOL: SCE

ABSTRACT

DURING SCHEDULED ISI INSPECTION, VISUAL EXAMINATION REVEALED
FAILED

ATTACHMENT WELDS ON TWO FEEDWATER SYSTEM PIPE SUPPORTS. AN
EXPANDED

INSPECTION OF THE SYSTEM DISCOVERED A TOTAL OF FOUR SUPPORT WELD
FAILURES. INITIAL SEISMIC ANALYSIS INDICATES SYSTEM STRUCTURAL
INTEGRITY WOULD BE MAINTAINED UNDER ORIGINAL STATION DESIGN
CRITERIA.

A SPECIFIC CAUSE OF FAILURE HAS NOT BEEN DETERMINED. A
METALLURGICAL

AND DESIGN ANALYSIS OF THESE SUPPORTS IS IN PROGRESS TO DETERMINE
THE

CAUSE OF FAILURE. ALL FAILED SUPPORTS ARE BEING REPLACED IN
ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE SECTION XI.

-----STEP: 1

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PI --INSTALLATION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:FI --CONDENSATE AND FEEDWATER
COMPONENT:SPT --SUPPORT
VENDOR:B130--BECHTEL CORP.
QUANTITY:4 --
TRAIN:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DA --BREAK/ SHEAR

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:NN --SEISMIC ACTIVITY
PRIMARY SYSTEM:FI --CONDENSATE AND FEEDWATER
SECONDARY SYSTEM:ZZ --UNKNOWN

COMPONENT:PICZ--PIPE, >6 AND <= 10 INCH DIAMETER, UNKNOWN
MATERI

QUANTITY:3 --

TRAIN:M --MULTIPLE

TIMING:P --POTENTIAL

PERFORMANCE:T --TOTAL FAULT

DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS

EFFECT:BP --EXTERNAL SYSTEM LEAKAGE, UNKNOWN RATE

INITIAL UNIT CONDITIONS: I COLD SHUTDOWN

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 6 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

213 1987 012 1 8807070550 209694 08/01/87

DOCKET:213 CONNECTICUT YANKEE TYPE:PWR

REGION: 1 NSSS:WE

ARCHITECTURAL ENGINEER: SWXX

FACILITY OPERATOR: CONN. YANKEE ATOMIC POWER CO.

SYMBOL: CYA

COMMENTS

STEP 4: MODEL 27 WITH INCONEL SB-166 TUBES. STEP 3: CAUSE EF - PRIMARY
WATER STRESS CORROSION CRACKING AND PITTING.

ABSTRACT

POWER LEVEL - 000%. SCHEDULED STEAM GENERATOR EDDY CURRENT
TESTING

(ECT) HAS BEEN CONDUCTED DURING THE CURRENT REFUELING OUTAGE IN
ACCORDANCE WITH ASME SECTION XI AND TECHNICAL SPECIFICATION 4.10.1,
"INSERVICE INSPECTION OF STEAM GENERATOR TUBES." ON AUGUST 1, 1987, IT
WAS DETERMINED THAT ECT DATA PLACED STEAM GENERATOR #1 INTO THE
C-3

CATEGORY. THE STEAM GENERATOR #1 INSPECTION PLAN WAS EXPANDED
FROM

456 TUBES TO 100 PERCENT (3,654 TUBES). ALSO, STEAM GENERATORS 2, 3,
AND 4 INSPECTION PLANS WERE EXPANDED TO INCLUDE TWO TIMES THE
ORIGINAL

SAMPLE SIZE. SUBSEQUENTLY, ON AUGUST 6, 1987, ECT DATA FOR STEAM

GENERATOR #2 PLACED IT IN CATEGORY C-3, THUS REQUIRING EXPANSION OF THE ECT PROGRAM TO 100 PERCENT OF THE TUBES IN ALL FOUR STEAM GENERATORS AS PROVIDED FOR BY TECHNICAL SPECIFICATIONS. AS A RESULT OF THE INSPECTION, A TOTAL OF 353 TUBES WERE PLUGGED. THE MAJOR CAUSE OF THE TUBE DEFECTS WAS ATTRIBUTED TO PRIMARY WATER STRESS CORROSION CRACKING AND, TO A LESSER EXTENT, PITTING. THIS EVENT IS REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(I) SINCE IT INVOLVES A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THIS REVISED REPORT IS BEING ISSUED TO PROVIDE ADDITIONAL INFORMATION ON THE IDENTIFIED CAUSES AND THE CORRECTIVE ACTIONS TAKEN.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PD --DESIGN ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
SUBLINK:A
CAUSE:ZZ --UNKNOWN
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:14 --
TRAIN:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DB --BEND/ DEFORM

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 1
SUBLINK:A
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:39 --

TRAIN:M --MULTIPLE
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 2 AND STEP 3

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:SG --STEAM GENERATOR
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:4 --
TRAIN:M --MULTIPLE
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

INITIAL UNIT CONDITIONS: I COLD SHUTDOWN
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 7 LER SCSS DATA 12-24-92

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
245	1987	020	1	8708190264	205751	06/26/87

DOCKET:245 MILLSTONE 1 TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: EBAS
FACILITY OPERATOR: NORTHEAST NUCLEAR ENERGY CO.
SYMBOL: NNE

ABSTRACT

POWER LEVEL - 000%. ON JUNE 26, 1987 AT 1010 HOURS, WHILE THE UNIT WAS SHUT DOWN FOR REFUELING, AN INDICATION WAS DISCOVERED ON CLASS I STAINLESS STEEL PIPING INSIDE THE DRYWELL. THE INDICATION WAS FOUND ON THE 22 INCH RECIRCULATION SYSTEM (AD) PIPE WELD RMBJ-1 (PIPE TO CAP

WELD) DURING INSERVICE INSPECTION (ISI) OF THE RECIRCULATION SYSTEM.
THE INDICATION WAS REVIEWED BY GENERATION MECHANICAL ENGINEERING

AGAINST THE ACCEPTANCE CRITERIA OF A.S.M.E. SECTION XI, AND ON JULY 1,
1987 WAS DECLARED AS BEING AN UNACCEPTABLE FLAW AND THUS
REPORTABLE.

THE WELD IN QUESTION WILL BE REPAIRED USING THE WELD OVERLAY
TECHNIQUE AND A PRESSURE TEST OF THE SYSTEM WILL BE PERFORMED.
THE

OVERLAY WAS COMPLETED AND A PRESSURE TEST WAS SATISFACTORILY
PERFORMED

ON AUGUST 9, 1987.

-----STEP: 1

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)

COMPONENT:PIES--PIPE, GREATER THAN 16 INCH DIAMETER, STAINLESS S

VENDOR:G080--GENERAL ELECTRIC CO.

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:DE --CRACK

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION

PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)

COMPONENT:PIES--PIPE, GREATER THAN 16 INCH DIAMETER, STAINLESS S

VENDOR:G080--GENERAL ELECTRIC CO.

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 8

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

245 1987 022 0 8708040266 205590 07/02/87

DOCKET:245 MILLSTONE 1 TYPE:BWR
 REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: EBAS
 FACILITY OPERATOR: NORTHEAST NUCLEAR ENERGY CO.
 SYMBOL: NNE

COMMENTS

WATCH LIST 975 - STRESS CORROSION CRACKING.

ABSTRACT

POWER EVEL - 000%. ON JULY 2, 1987 AT 1500 HOURS, WHILE THE UNIT WAS
SHUT DOWN FOR REFUELING, INSERVICE INSPECTION ULTRASONIC TESTING
IDENTIFIED UNACCEPTABLE INCLUSIONS IN THE (BL) RETURN LINE PIPING.
THE INCLUSIONS ARE IN THE INSPECTION ZONE OF WELD ICBC-F-14 ON THE 8"
STAINLESS STEEL RETURN LINE AND EXTEND AXIALLY FOR APPROXIMATELY

26

INCHES IN THE PIPING MATERIAL. THE INCLUSIONS WERE REVIEWED BY
GENERATION MECHANICAL ENGINEERING AGAINST THE ACCEPTANCE
CRITERIA OF

ASME SECTION XI AND WAS DECLARED AS BEING AN UNACCEPTABLE FLAW
AND

THUS REPORTABLE. THE PIPE IN QUESTION WAS REPLACED USING A
MATERIAL

THAT IS MORE RESISTANT TO INTERGRANULAR STRESS CORROSION CRACKING
AND

A SYSTEM HYDROSTATIC TEST WILL BE PERFORMED.

-----STEP: 1

 CAUSE:EF --CORROSION/ OXIDATION

 PRIMARY SYSTEM:BB --ISOLATION CONDENSER (BWR)

 COMPONENT:PICS--PIPE, >6 AND <= 10 INCH DIAMETER, STAINLESS STEE

 VENDOR:D240--DRAVO, INC.

 TIMING:M --ACTUAL PREEXISTING, UNDETECTED

 PERFORMANCE:PR --PARTIAL FAULT/REPAIR

 DETECTION:I --TESTING/ INSPECTION

 EFFECT:DE --CRACK

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

 SUBLINK:X

 CAUSE:PA --WELD-RELATED FLAW

 PRIMARY SYSTEM:BB --ISOLATION CONDENSER (BWR)

 COMPONENT:PICS--PIPE, >6 AND <= 10 INCH DIAMETER, STAINLESS STEE

 VENDOR:D240--DRAVO, INC.

 TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 11 LER SCSS DATA 12-24-92

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
259	1981	037	3	8311090350	187466	06/29/81

DOCKET:259 BROWNS FERRY 1 TYPE:BWR
REGION: 2 NSSS:GE
ARCHITECTURAL ENGINEER: TVAX
FACILITY OPERATOR: TENNESSEE VALLEY AUTHORITY
SYMBOL: TVA

COMMENTS
IE BULLETIN 83-02

ABSTRACT

REACTOR WATER CLEANUP SYSTEM INBOARD PRIMARY CONTAINMENT
ISOLATION
VALVE 69-1 WAS ULTRASONICALLY INSPECTED ON 6/21/81. AN INDICATION WAS

OBSERVED ON THE VESSEL SIDE OF THE VALVE WELD. (TECH SPEC 3.6.G.) A
SUBSEQUENT RADIOGRAPH SHOWED NO INDICATIONS. ON 6/24/81 A UT
INSPECTION BY A LEVEL II INSPECTOR CONFIRMED AN INDICATION. PREVIOUS

SIMILAR EVENT: BFRO-50-259/80-004. FOR CAUSE SEE ATTACHED
METALLURGICAL REPORT. THE VELAN MODEL 900 LB. 6 INCH. (SS) STAINLESS
STEEL GATE VALVE, RATED AT 1146 PSI AT 500 F (NORMAL OPERATING
PRESSURE 960 PSI) WAS REPLACED. THE INDICATION IS ACCEPTABLE UNDER
THE SUMMER 1978 ADDENDA TO ASME SECTION XI CODE. THIS IS CONSIDERED
A
RANDOM EVENT.

-----STEP: 1

CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION

SECONDARY SYSTEM:WP --REACTOR WATER CLEANUP (BWR)
COMPONENT:ISVS--VALVE, ISOLATION/ SHUTOFF, STAINLESS STEEL
VENDOR:V085--VELAN VALVE CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BK --EXTERNAL SYSTEM LEAKAGE, SMALL

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION

PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION

SECONDARY SYSTEM:WP --REACTOR WATER CLEANUP (BWR)

COMPONENT:ISVS--VALVE, ISOLATION/ SHUTOFF, STAINLESS STEEL

VENDOR:V085--VELAN VALVE CORP.

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:BK --EXTERNAL SYSTEM LEAKAGE, SMALL

INITIAL UNIT CONDITIONS: H REFUELING

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 16 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

266 1984 002 0 8404110360 189150 02/28/84

DOCKET:266 POINT BEACH 1 TYPE:PWR

REGION: 3 NSSS:WE

ARCHITECTURAL ENGINEER: BECH

FACILITY OPERATOR: WISCONSIN ELECTRIC POWER CO.

SYMBOL: WEP

ABSTRACT

POWER LEVEL - 000%. DURING UNIT 1 REFUELING 11, AN INSPECTION OF THE REACTOR VESSEL OUTLET NOZZLE-TO-SHELL WELDS WAS PERFORMED USING PROGRAMMED AND REMOTE MECHANIZED ULTRASONICS. THIS EXAMINATION WAS A

PART OF THE FIRST PERIOD INSERVICE INSPECTION FOR THE SECOND
TEN-YEAR
INTERVAL. FOUR INDICATIONS IN THE "A" NOZZLE AND SEVEN INDICATIONS IN

THE "C" NOZZLE WERE RECORDED, SIZED, AND EVALUATED. ON 02/28/84, THE
LICENSEE WAS NOTIFIED THAT THERE WAS ONE INDICATION IN EACH OF THE
OUTLET NOZZLE WELDS THAT EXCEEDED THE CODE ALLOWABLE SIZE. THESE

INDICATIONS ARE LOCATED IN THE ROOT AREA OF THE WELDS AND ARE
BELIEVED, BY THE EVALUATOR, TO BE SLAG. THE LICENSEE'S CONSULTANT
(SWRI) HAS PERFORMED BEAM SPREAD EXPERIMENTS ON THE POINT BEACH
CALIBRATION BLOCK AND FRACTURE ANALYSIS OF THE FLAW INDICATIONS.
FINAL RESULTS FROM THESE STUDIES BY SWRI INDIVIDUALLY DEMONSTRATE
THAT

THE INDICATIONS MEET THE ACCEPTANCE CRITERIA OF ASME SECTION XI,
THEREBY CONFIRMING THE CONTINUED SERVICEABILITY OF THE REACTOR
VESSEL.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PF --FABRICATION ACTIVITY
COMPONENT:PCP --CONTRACTOR PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:AF --PRESSURIZER (PWR)
COMPONENT:NZL --NOZZLE
VENDOR:B015--BABCOCK & WILCOX COMPANY
QUANTITY:2 --
TRAIN:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 30 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
328 1984 012 0 8408160269 191063 07/10/84

DOCKET:328 SEQUOYAH 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: TVAX
FACILITY OPERATOR: TENNESSEE VALLEY AUTHORITY
SYMBOL: TVA

ABSTRACT

POWER LEVEL - 100%. DURING SURVEILLANCE TESTING FOR EXTERNAL (TO
CONTAINMENT) PIPING LEAKAGE, BOTH TRAINS OF THE RESIDUAL HEAT
REMOVAL

SYSTEM WERE INOPERABLE FOR 2 HRS, 47 MINS ON 7-10-84 WHEN VALVE
HCV-74-34 (RHR TO RWST RECIRC LINE ISOLATION VALVE) WAS OPENED AS
PART

OF THE PROCEDURE FOR CHECKING RHR PIPE LEAKAGE. REVIEW OF THE RHR

SYSTEM DETERMINED THAT THE LEAKAGE INSPECTION COULD BE SATISFIED
WITH

THE RWST HEAD PRESSURE ON THE RHR-RWST RECIRC LINE AND HCV-74-34
NEED

NOT BE OPENED. A PROCEDURE CHANGE WAS ISSUED TO DELETE THE
REQUIREMENT TO OPEN HCV-74-34, AND THE INSPECTION WAS SATISFACTORILY

COMPLETED ON 7-11-84. ALL SIMILAR INSTRUCTIONS PERTAINING TO SECTION
XI EXTERNAL LEAKAGE INSPECTIONS ARE BEING REVIEWED TO ENSURE
ADDITIONAL IMPROPER VALVE LINEUPS ARE NOT BEING REQUIRED.

-----STEP: 1

CAUSE:SB --TASK DESCRIPTION INADEQUACY
PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:BF --RESIDUAL HEAT REMOVAL (PWR)
SECONDARY SYSTEM:CD --BORATED/ REFUELING WATER STORAGE (PWR)

COMPONENT:ISVZ--VALVE, ISOLATION/ SHUTOFF, UNKNOWN MATERIAL
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:AK --TRANSFER OPEN

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:RT --RESULTANT TOTAL SYSTEM FAULT
PRIMARY SYSTEM:BF --RESIDUAL HEAT REMOVAL (PWR)
COMPONENT:XXX --ENTIRE SYSTEM
TIMING:P --POTENTIAL
PERFORMANCE:T --TOTAL FAULT
DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:KB --FAILURE TO OPERATE

INITIAL UNIT CONDITIONS: E STEADY-STATE OPERATION
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 32 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
333 1984 021 0 8411190152 192246 10/12/84

DOCKET:333 FITZPATRICK TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: POWER AUTHORITY OF THE STATE OF NY
SYMBOL: PNY

ABSTRACT

POWER LEVEL - 000%. INSERVICE INSPECTION PERFORMED IN CONJUNCTION WITH THE APPLICATION OF INDUCTION HEATING STRESS IMPROVEMENT (IHSI) REVEALED INDICATIONS OF INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) IN SEVEN 12 INCH DIAMETER RECIRCULATION RISER WELDS AND IN FOUR 28 INCH DIAMETER RECIRCULATION SUCTION PIPE WELDS. FIVE OF THE

12 INCH PIPING JOINTS WERE REPAIRED BY WELD OVERLAYS WHICH PROVIDED FULL STRUCTURAL DESIGN MARGINS FOR THIS PIPING. THE REMAINING SIX

WELDS WERE EVALUATED IN ACCORDANCE WITH SECTION XI 1983 EDITION, ASME BOILER AND PRESSURE VESSEL CODE AS MODIFIED BY NRC GENERIC LETTER 84-11. IN ALL SIX CASES, THE INDICATIONS WERE SHOWN NOT TO AFFECT THE DESIGN STRUCTURAL MARGINS OF THIS PIPING FOR LENGTHS OF TIME EXCEEDING THAT PERIOD UNTIL THE NEXT SCHEDULED REFUELING OUTAGE.

ADDITIONALLY, THE APPLICATION OF IHSI ON THESE WELDS SHOULD FURTHER INHIBIT CRACK PROPAGATION THEREBY INCREASING THEIR ACCEPTABILITY FOR FUTURE OPERATION WITHOUT REPAIR. BOTH THE JOINTS REPAIRED BY WELD OVERLAY AND THOSE WITH UNREPAIRED FLAWS HAVE BEEN SHOWN TO MEET THE REQUIREMENTS OF THE JAFNPP TECH SPECS, 4.6.F. DETAILED WELD BY WELD INFORMATION CONCERNING EXAMINATION, REPAIR AND STRUCTURAL ANALYSIS WAS PROVIDED TO THE DIRECTOR OF NRR ON 10-21-84 IN LETTER JAFP 84-0979.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PC --CONSTRUCTION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:CE --WEAR/ AGE/ END OF LIFE
PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)
COMPONENT:PIDZ--PIPE, >10 AND <=16 INCH DIAMETER, UNKNOWN MATERI
VENDOR:K055--KELLOGG, M.W. CO., THE
QUANTITY:7 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2
SUBLINK:X
CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)

COMPONENT:PIDZ--PIPE, >10 AND <=16 INCH DIAMETER, UNKNOWN
MATERI

VENDOR:K055--KELLOGG, M.W. CO., THE
QUANTITY:7 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:CE --WEAR/ AGE/ END OF LIFE
PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)
COMPONENT:PIEZ--PIPE, GREATER THAN 16 INCH DIAMETER, UNKNOWN
MAT

VENDOR:K055--KELLOGG, M.W. CO., THE
QUANTITY:4 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 4

SUBLINK:X
CAUSE:PA --WELD-RELATED FLAW
PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)
COMPONENT:PIEZ--PIPE, GREATER THAN 16 INCH DIAMETER, UNKNOWN
MAT

VENDOR:K055--KELLOGG, M.W. CO., THE
QUANTITY:4 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: Z UNKNOWN/ NOT APPLICABLE
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 44

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

373 1986 019 0 8606240625 199782 05/21/86

DOCKET:373 LA SALLE 1 TYPE:BWR
REGION: 3 NSSS:GE
ARCHITECTURAL ENGINEER: SLXX
FACILITY OPERATOR: COMMONWEALTH EDISON CO.
SYMBOL: CWE

COMMENTS

STEP 1: CAUSE PA- SLAG OR OTHER DEFECT IN A PLUG-TYPE REPAIR WELD.

ABSTRACT

POWER LEVEL - 000%. TWO THROUGH WALL LEAKS WERE IDENTIFIED DURING THE

ASME SECTION XI LEAKAGE TEST ON THE CLASS 1 PORTION OF THE RESIDUAL HEAT REMOVAL (RHR) "B" AND "C" LOW PRESSURE COOLANT INJECTION (LPCI) LOOPS. THE LEAKAGE WAS COMING FROM PRESERVICE PLUG-TYPE REPAIR WELDS

IN THE VICINITY OF THE 1E12-F092B AND 1E12-F092C VALVES. THE LEAKING PLUGS WILL BE REPAIRED DURING THE PRESENT UNIT 1 REFUEL OUTAGE.

THE

UNIT WAS IN COLD SHUTDOWN.

-----STEP: 1

SUBLINK:A

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:BH --RESIDUAL HEAT REMOVAL (BWR)

SECONDARY SYSTEM:SC --REACTOR DRYWELL (BWR)

COMPONENT:PIZC--PIPE, UNKNOWN DIAMETER, CARBON STEEL

QUANTITY:2 --

TRAIN:M --MULTIPLE

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:BK --EXTERNAL SYSTEM LEAKAGE, SMALL

-----STEP: 2

SUBLINK:A

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:BH --RESIDUAL HEAT REMOVAL (BWR)

SECONDARY SYSTEM:SC --REACTOR DRYWELL (BWR)

COMPONENT:PIZC--PIPE, UNKNOWN DIAMETER, CARBON STEEL

DIFFERENTIATOR:2 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:F --OPERATIONAL ABNORMALITY

EFFECT:BP --EXTERNAL SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 1 AND STEP 2

INITIAL UNIT CONDITIONS: I COLD SHUTDOWN

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 1

EFFECT ON ENVIRONMENT: R RADIOLOGICAL RELEASE TO CONTAINMENT

EFFECT ON PERSONNEL: N NO EXPOSURE

-----STEP: 5

INITIAL UNIT CONDITIONS:

UNIT EFFECT: HA AFFECTED OTHER UNIT(S)

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 5

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:BH --RESIDUAL HEAT REMOVAL (BWR)

SECONDARY SYSTEM:SC --REACTOR DRYWELL (BWR)

COMPONENT:PIZC--PIPE, UNKNOWN DIAMETER, CARBON STEEL

QUANTITY:3 --

TRAIN:M --MULTIPLE

DIFFERENTIATOR:3 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:F --OPERATIONAL ABNORMALITY

EFFECT:BP --EXTERNAL SYSTEM LEAKAGE, UNKNOWN RATE

INITIAL UNIT CONDITIONS: Z UNKNOWN/ NOT APPLICABLE

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 45

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

387 1982 001 1 8305120430 183927 07/18/82

DOCKET:387 SUSQUEHANNA 1 TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: PENNSYLVANIA POWER & LIGHT CO.
SYMBOL: PPL

ABSTRACT

WHILE PERFORMING OFFSET RADIOGRAPHY OF THE RECIRC RISER TO SWEEPOLET/REDUCER WELDS AS A SUPPLEMENTAL EXAMINATION TO THE PSI ULTRASONIC DATA, LINEAR INDICATIONS WERE FOUND ON A RECIRC RISER WELD;

A-11M. ADDITIONAL RADIOGRAPHY WAS PERFORMED ON ALL THE RECIRC RISERS

WELDS AND ONE ADDITIONAL INDICATION WAS FOUND ON THE A-13 WELD. THE

EXTENT OF THE INDICATIONS DID NOT POSTULATE PROBABLE FAILURE. THE CAUSE OF THE LINEAR INDICATIONS WAS SLAG INCLUSION IN THE A-11M WELD

AND SLAG INCLUSION AND PARTIAL COLD LAP (1/4") OF THE A-13 WELD. THE AFFECTED WELD AREAS WERE EXCAVATED (APPROX. 6" ON A-11M AND APPROX. 3"

ON A-13) AND WELD REPAIRED WITH A PP&L APPROVED REPAIR PLAN IN ACCORDANCE WITH ASME SECTION XI B&PV CODE. ALL RISER TO SWEEPOLET/REDUCER WELDS HAVE BEEN ACCEPTED. NO FUTURE ACTIONS ARE PLANNED.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR
PRIMARY SYSTEM:PI --INSTALLATION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:AI --RECIRCULATING WATER (BWR)

COMPONENT:PICZ--PIPE, >6 AND <= 10 INCH DIAMETER, UNKNOWN MATERI

VENDOR:B130--BECHTEL CORP.

QUANTITY:2 --

TRAIN:Z --UNKNOWN

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:P --PARTIAL FAULT

DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 48 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
414 1987 012 0 8705010010 204261 02/26/87

DOCKET:414 CATAWBA 2 TYPE:PWR
REGION: 2 NSSS:WE
ARCHITECTURAL ENGINEER: DUKE
FACILITY OPERATOR: DUKE POWER CO.
SYMBOL: DPC

COMMENTS
STEP 2: MODEL 306.

ABSTRACT

POWER LEVEL - 000%. ON FEBRUARY 26, 1987, AN INOPERABLE SNUBBER WAS FOUND DURING REMOVAL AND REPLACEMENT ACTIVITIES. THE REMOVAL AND REPLACEMENT WORK HAD BEEN INITIATED TO PROVIDE TECHNICIANS ACCESS TO

THE ACTUATOR OF A STEAM GENERATOR BLOWDOWN CONTAINMENT ISOLATION

BYPASS VALVE, WHICH HAD FAILED TO MEET ITS STROKE TIME DURING ASME

SECTION XI, SUBSECTION IWV, VALVE INSERVICE TESTING PROGRAM, QUARTERLY

TESTING. ON FEBRUARY 26, AN ATTEMPT WAS MADE TO STROKE THE SNUBBER

PRIOR TO REINSTALLATION, BUT THE SNUBBER COULD NOT BE CYCLED. A NEW

SNUBBER WAS INSTALLED AT THIS POINT. ON MARCH 24, 1987, AT 0900 HOURS IT WAS FOUND THAT THE REQUIRED ENGINEERING EVALUATION WAS NOT PERFORMED WHICH RESULTED IN A TECHNICAL SPECIFICATION VIOLATION. THE

UNIT WAS IN MODES 1, POWER OPERATION, 2, STARTUP, AND 3, HOT STANDBY,

THROUGHOUT THIS INCIDENT. THIS INCIDENT HAS BEEN CLASSIFIED AS EVENT

CAUSE CODE D, DEFECTIVE PROCEDURES. THE CONTROLLING PROCEDURE FOR

SUPPORT/RESTRAINT MAINTENANCE ACTIVITIES DID NOT INCLUDE INSTRUCTIONS

FOR NOTIFICATION OF THE APPROPRIATE DUKE POWER PERSONNEL IN THE EVENT

AN INOPERABLE SNUBBER IS FOUND DURING REMOVAL AND REPLACEMENT MAINTENANCE ACTIVITIES. THE SUBJECT PROCEDURE WILL BE REVISED TO PREVENT SIMILAR INCIDENTS. AT THE TIME THE INOPERABILITY WAS DISCOVERED ON MARCH 24, DESIGN ENGINEERING WAS NOTIFIED AND AN ENGINEERING EVALUATION WAS REQUESTED.

-----STEP: 1

SUBLINK:A

CAUSE:ZZ --UNKNOWN

PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION

SECONDARY SYSTEM:WF --STEAM GENERATOR BLOWDOWN (PWR)

COMPONENT:BYVZ--VALVE, BYPASS, UNKNOWN MATERIAL

TIMING:G --ACTUAL PREEXISTING, DETECTED

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:KE --LONG RESPONSE TIME

-----STEP: 2

CAUSE:ZZ --UNKNOWN

PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION

SECONDARY SYSTEM:WF --STEAM GENERATOR BLOWDOWN (PWR)

COMPONENT:SNB --SNUBBER

VENDOR:P029--PACIFIC SCIENTIFIC COMPANY

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:KB --FAILURE TO OPERATE

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:SB --TASK DESCRIPTION INADEQUACY

PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY

COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:T --TOTAL FAULT

DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS

EFFECT:TC --OMISSION WITHIN ALLOTTED TIME

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 3
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:WF --STEAM GENERATOR BLOWDOWN (PWR)
COMPONENT:BYVZ--VALVE, BYPASS, UNKNOWN MATERIAL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:M --NO FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:PL --TEST NOT PERFORMED

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 3
SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:WF --STEAM GENERATOR BLOWDOWN (PWR)
COMPONENT:PIZZ--PIPE, UNKNOWN DIAMETER, UNKNOWN MATERIAL
QUANTITY:Z --UNKNOWN
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:M --NO FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:PL --TEST NOT PERFORMED

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 1 AND STEP 5

INITIAL UNIT CONDITIONS: L HOT STANDBY
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 50 LER SCSS DATA 12-24-92

DOCKET	YEAR	LER	NUMBER	REVISION	DCS	NUMBER	NSIC	EVENT	DATE
528	1985	025	0	8507160331	197941	03/14/85			

DOCKET:528 PALO VERDE 1 TYPE:PWR
REGION: 5 NSSS:CE
ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: ARIZONA PUBLIC SERVICE CO.
SYMBOL: APS

COMMENTS

OTHER REPORTABILITY - VOLUNTARY REPORT; STEP 5: EFFECT KX - SMALL REACTIVITY ADDITIONS BY WITHDRAWING RODS.

ABSTRACT

POWER LEVEL - 000%. ON MARCH 14, 1985, TECHNICAL SPECIFICATION (TS) 3.1.2.3 WAS DEVIATED FROM BY USING THE TRAIN "A" HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM TO PROVIDE THE REQUIRED BORON INJECTION FLOWPATH WHEN THE ASME SECTION XI SURVEILLANCE TEST WAS NOT CURRENT ON

THE TRAIN "A" HPSI PUMP. DURING THIS TIME, PORTIONS OF THE CONTROL ELEMENT ASSEMBLY (CEA) PATCH VERIFICATION TESTING WERE PERFORMED RESULTING IN SMALL INCREMENTS OF POSITIVE REACTIVITY ADDITION. AS SOON AS THE LACK OF A CURRENT SURVEILLANCE TEST WAS REALIZED, THE PATCH VERIFICATION TESTING WAS STOPPED, THE ASME SECTION XI TEST FOR

TRAIN "A" HPSI WAS PERFORMED, AND THE PUMP AND ASSOCIATED BORON INJECTION FLOWPATH WAS DECLARED OPERABLE. THE OPERATOR PERFORMING THE

BORATION FLOWPATH SURVEILLANCE TEST (41ST-1CH02) BELIEVED THAT THE REQUIRED ASME SECTION XI TESTS ON THE HPSI PUMP HAD BEEN SUCCESSFULLY

COMPLETED. THE EMERGENCY BORATION PROCEDURE HAS BEEN REVIEWED AND

MODIFIED AS NECESSARY TO PROVIDE IMPROVED GUIDANCE TO THE OPERATIONS

STAFF ON EMERGENCY BORATION WHILE IN MODE 5. IN ADDITION, UNIT 1 PERSONNEL HAVE BEEN INSTRUCTED TO ENSURE THAT THEY MUST VERIFY STEPS

IN THE PROCEDURE AND NOT RELY ON MEMORY OF COMPLETED SURVEILLANCE TESTING.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR

PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY

COMPONENT:PNO --NON-LICENSED OPERATOR. PERSONNEL

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:T --TOTAL FAULT

DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS

EFFECT:TC --OMISSION WITHIN ALLOTTED TIME

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:RC --RESULTANT COMPONENT FAULT

PRIMARY SYSTEM:BL --INTERMEDIATE PRESSURE INJECTION (PWR)

COMPONENT:PMPZ--PUMP, UNKNOWN TYPE

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:M --NO FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:PL --TEST NOT PERFORMED

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

SUBLINK:A
CAUSE:RS --RESULTANT SUBSYSTEM FAULT
PRIMARY SYSTEM:BL --INTERMEDIATE PRESSURE INJECTION (PWR)
COMPONENT:1XZ --ONE OUT OF AN UNKNOWN NUMBER OF TRAINS
TIMING:P --POTENTIAL
PERFORMANCE:T --TOTAL FAULT
DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:HI --LOW FLOW

-----STEP: 4

CAUSE:SE --PROPER HUMAN ACTION OR RESPONSE
PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY
COMPONENT:PLO --LICENSED OPERATOR PERSONNEL
DIFFERENTIATOR:2 --
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 4

SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AA --REACTOR CORE
COMPONENT:CRA --CONTROL ROD ASSEMBLY
QUANTITY:Z --UNKNOWN
TRAIN:Z --UNKNOWN
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:M --NO FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:KX --OTHER (FUNCTIONAL)

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 3 AND STEP 5

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AA --REACTOR CORE
COMPONENT:FLA --FUEL ASSEMBLY
QUANTITY:A --
TIMING:P --POTENTIAL
PERFORMANCE:T --TOTAL FAULT

DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:PE --FLUX ANOMALY

INITIAL UNIT CONDITIONS: I COLD SHUTDOWN
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 52 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
245 1989 009 0 8905190082 214499 04/20/89

DOCKET:245 MILLSTONE 1 TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: EBAS
FACILITY OPERATOR: NORTHEAST NUCLEAR ENERGY CO.
SYMBOL: NNE

ABSTRACT

POWER LEVEL - 000%. ON 4/20/89 AT 1135 HOURS, WHILE THE UNIT WAS SHUT
DOWN FOR REFUELING, AN INDICATION WAS DISCOVERED ON CLASS I
STAINLESS

STEEL PIPING INSIDE THE DRYWELL. THE INDICATION WAS FOUND ON THE 8"
REACTOR WATER CLEAN-UP (CE) PIPE WELD CUAJ-5 (PIPE TO VALVE 1-CU-2
WELD) DURING INSERVICE INSPECTION (ISI) OF THE REACTOR WATER
CLEAN-UP

SYSTEM. THE INDICATION WAS REVIEWED AGAINST THE ACCEPTANCE
CRITERIA

OF ASME SECTION XI, AND ON 4/22/89 WAS DECLARED AS BEING AN
UNACCEPTABLE FLAW AND THUS REPORTABLE. THE WELD IN QUESTION WAS
REPAIRED USING THE WELD OVERLAY TECHNIQUE, AND A HYDROSTATIC TEST
WAS
PERFORMED.

-----STEP: 1

CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:WP --REACTOR WATER CLEANUP (BWR)
COMPONENT:PICS--PIPE, >6 AND <= 10 INCH DIAMETER, STAINLESS STEE
VENDOR:E065--EBASCO SERVICES INC.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION

EFFECT:DE --CRACK

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:X

CAUSE:PA --WELD-RELATED FLAW

PRIMARY SYSTEM:WP --REACTOR WATER CLEANUP (BWR)

COMPONENT:PICS--PIPE, >6 AND <= 10 INCH DIAMETER, STAINLESS STEE

VENDOR:E065--EBASCO SERVICES INC.

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:DE --CRACK

INITIAL UNIT CONDITIONS: H REFUELING

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 55

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

263 1989 033 0 8911280380 215948 10/25/89

DOCKET:263 MONTICELLO TYPE:BWR

REGION: 3 NSSS:GE

ARCHITECTURAL ENGINEER: BECH

FACILITY OPERATOR: NORTHERN STATES POWER CO.

SYMBOL: NSP

COMMENTS

STEP 7: COMP X - ESF TYPE: ACTUATION OF TWO SAFETY RELIEF VALVES.

ABSTRACT

POWER LEVEL - 000%. ON 10/25/89, WITH THE PLANT IN COLD SHUTDOWN,
DURING PERFORMANCE OF THE SECTION XI ASME BOILER AND PRESSURE
VESSEL

CODE REACTOR COOLANT SYSTEM LEAKAGE TEST, INITIATION OF DIVISION II
LOW LOW SET LOGIC RESULTED IN THE OPERATION OF SAFETY RELIEF
VALVES

"G" AND "H". THE INITIATION SIGNAL OCCURRED AS A RESULT OF AN
INSTRUMENT SENSING LINE PRESSURE SPIKE CAUSED BY SEATING OF AN
EXCESS

FLOW CHECK VALVE DURING TESTING. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE TESTING PROCEDURES. INVESTIGATION AND EVALUATION OF THE SAFETY RELIEF VALVE LINES, AND THE TEST PROCEDURE REVISION WERE DONE PRIOR TO RECOMMENCING THE VESSEL LEAKAGE TEST. REVIEW OF ALL EXCESS FLOW CHECK VALVE TEST PROCEDURES FOR PROPER ISOLATIONS WILL BE COMPLETED BEFORE THE NEXT SCHEDULED REFUELING OUTAGE. A REVIEW OF THIS EVENT WILL BE INCLUDED IN TECHNICAL STAFF CONTINUING TRAINING.

-----STEP: 1

CAUSE:SB --TASK DESCRIPTION INADEQUACY
PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:TA --OMISSION OF TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:IW --ENGINEERED SAFETY FEATURES ACTUATION
SECONDARY SYSTEM:BR --NUCLEAR BOILER OVERPRESSURE PROTECTION (BWR)
COMPONENT:HS --SWITCH, HAND
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:AM --WRONG POSITION

-----STEP: 3

CAUSE:SE --PROPER HUMAN ACTION OR RESPONSE
PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
DIFFERENTIATOR:2 --
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:M --NO FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UF --COMMISSION OF A DESIRED TASK, ANALYSIS, OR STEP

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 3

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:BR --NUCLEAR BOILER OVERPRESSURE PROTECTION (BWR)

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:M --NO FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:AP --TRANSFER CLOSED

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 4

SUBLINK:A

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:IW --ENGINEERED SAFETY FEATURES ACTUATION
SECONDARY SYSTEM:BR --NUCLEAR BOILER OVERPRESSURE PROTECTION
(BWR)

COMPONENT:SL --SENSING LINE
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:HK --HIGH STATIC PRESSURE

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 2 AND STEP 5

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:IW --ENGINEERED SAFETY FEATURES ACTUATION
SECONDARY SYSTEM:BR --NUCLEAR BOILER OVERPRESSURE PROTECTION
(BWR)

COMPONENT:PT --TRANSMITTER, PRESSURE
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:LE --ERRONEOUS/ NO SIGNAL

-----STEP: 7

THIS STEP IS DIRECTLY LINKED TO STEP 6

CAUSE:VW --PROCEDURAL DEFICIENCY

INITIAL UNIT CONDITIONS: I COLD SHUTDOWN
UNIT EFFECT: EB UNINTENTIONAL AUTOMATIC ESF ACTUATION

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
213 1989 017 1 9007300002 218985 09/25/89

DOCKET:213 CONNECTICUT YANKEE TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: CONN. YANKEE ATOMIC POWER CO.
SYMBOL: CYA

COMMENTS

STEPS 2,5,8: CAUSE EF - PRIMARY WATER STRESS CORROSION CRACKING. STEPS

2,3,5,6,8,9: INCONEL 600 SB-166. STEPS 4,7,10: MODEL 27.

ABSTRACT

POWER LEVEL - 000%. SCHEDULED STEAM GENERATOR EDDY CURRENT
TESTING

(ECT) HAS BEEN CONDUCTED DURING THE CURRENT REFUELING OUTAGE IN
ACCORDANCE WITH ASME SECTION XI AND TECHNICAL SPECIFICATION 4.10.1,
"INSERVICE INSPECTION OF STEAM GENERATOR TUBES." ON SEPTEMBER 25,
1989, AT APPROXIMATELY 1330 HOURS, WITH THE PLANT IN MODE 6, IT WAS
DETERMINED THAT FINAL ECT DATA PLACED STEAM GENERATOR #2 INTO THE
C-3

CATEGORY. AS PLANNED, 100% OF STEAM GENERATOR #2 TUBES (3493 TUBES)
WERE INSPECTED. SUBSEQUENTLY, ON OCTOBER 4, 1989 FINAL ECT RESULTS
FOR NO. 4 STEAM GENERATOR PLACED IT IN CATEGORY C-3, THUS REQUIRING

EXPANSION OF THE ECT PROGRAM TO 100 PERCENT OF THE TUBES IN ALL
FOUR

STEAM GENERATORS AS ORIGINALLY PLANNED. THE MAJOR CAUSE OF THE
TUBE

DEFECTS WAS ATTRIBUTED TO PRIMARY WATER STRESS CORROSION
CRACKING AND,

TO A LESSER EXTENT, PITTING. AS A RESULT OF THE INSPECTION, A TOTAL
OF 159 TUBES WERE PLUGGED. THIS EVENT IS REPORTABLE PER
10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE
PLANT'S TECHNICAL SPECIFICATIONS. THIS SUPPLEMENTAL REPORT IS BEING
ISSUED TO PROVIDE ADDITIONAL INFORMATION ON THE IDENTIFIED CAUSES
AND

CORRECTIVE ACTION TAKEN.

-----STEP: 1

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PZ --UNKNOWN ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL

TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
QUANTITY:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2
SUBLINK:X
CAUSE:FA --EROSION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
QUANTITY:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 2
SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:SG --STEAM GENERATOR
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 1
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
QUANTITY:M --MULTIPLE
TRAIN:2 --

DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 5

SUBLINK:X
CAUSE:FA --EROSION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
QUANTITY:M --MULTIPLE
TRAIN:2 --
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 7

THIS STEP IS DIRECTLY LINKED TO STEP 5

SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:SG --STEAM GENERATOR
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
TRAIN:2 --
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 8

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
QUANTITY:M --MULTIPLE
TRAIN:3 --
DIFFERENTIATOR:3 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:PR --PARTIAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 9

THIS STEP IS DIRECTLY LINKED TO STEP 8

SUBLINK:X

CAUSE:FA --EROSION

PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)

COMPONENT:TBG --TUBING

QUANTITY:M --MULTIPLE

TRAIN:3 --

DIFFERENTIATOR:3 --

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 10

THIS STEP IS DIRECTLY LINKED TO STEP 8

SUBLINK:A

CAUSE:RC --RESULTANT COMPONENT FAULT

PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)

COMPONENT:SG --STEAM GENERATOR

VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.

TRAIN:3 --

DIFFERENTIATOR:3 --

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:PR --PARTIAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

-----STEP: 11

THIS STEP IS DIRECTLY LINKED TO STEP 4 AND STEP 7
AND STEP 10

INITIAL UNIT CONDITIONS: H REFUELING

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 58

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

255 1989 025 0 8912280118 216257 11/21/89

DOCKET:255 PALISADES

TYPE:PWR

REGION: 3

NSSS:CE

ARCHITECTURAL ENGINEER: BECH
FACILITY OPERATOR: CONSUMERS POWER CO.
SYMBOL: CPC

COMMENTS

STEP 2: MODEL NO. 12011(CF8M) FJMN PQTYZ. STEPS 3,4: MODEL NO. 88RR-001.

ABSTRACT

POWER LEVEL - 000%. ON 11/21/89 AT 0225 POWER OPERATED RELIEF VALVE (PORV), PRV-1042B (AB;RV) INSTANTANEOUSLY OPENED FOLLOWING THE MANUAL OPENING OF ITS ASSOCIATED MOTOR OPERATED BLOCK VALVE (MOV), MO-10242 (AB;ISV). BOTH MOV AND PORV, AND THE REDUNDANT VALVES MO-1043A AND

PRV-1043B, HAD RECENTLY BEEN REPLACED DURING ONGOING MAINTENANCE

OUTAGE. THE MOV WAS OPENED WITH THE REACTOR SUBCRITICAL AND THE PRIMARY COOLANT SYSTEM (PCS) AT 2154 PSIA TO PERMIT ASME SECTION XI LEAK TESTING OF THE WELDS MADE DURING VALVE INSTALLATION. APPROX.

24

SECS FOLLOWING THE OPENING OF THE PORV, THE REACTOR AUTOMATICALLY TRIPPED DUE TO THE RECEIPT OF ALL 4 THERMAL MARGIN LOW PRESSURE (TMLP)

SIGNALS. APPROX. 3 MINS LATER A SAFETY INJECTION ACTUATION SIGNAL (SIAS) WAS RECEIVED WHEN PCS PRESSURE DROPPED TO THE SAFETY INJECTION

SYSTEM ACTUATION PRESSURE OF 1605 PSIA. ALL SAFETY INJECTION EQUIPMENT THEN STARTED OR OTHERWISE PERFORMED ITS DESIGN FUNCTION.

BOTH DIESEL GENERATORS STARTED ON LOW BUS VOLTAGE, BUT NO EQUIPMENT

WAS AUTOMATICALLY LOADED ONTO THE DIESEL GENERATORS. WITH THE PCS AT

1565 PSIA THE PORV CLOSED AND THE MOV FULLY CLOSED. AN UNUSUAL EVENT

WAS DECLARED DUE TO THE PORV OPENING AT 0256. THE PORV AND THE MOV

WHICH OPENED WERE REMOVED FROM THE PLANT AND INSPECTED AND TESTED.

THE VALVE OPERATING CHARACTERISTICS WERE DETERMINED, AND THE PLANT

OPERATORS TRAINED AND OPERATING PROCEDURES MODIFIED TO REFLECT THOSE

CHARACTERISTICS.

---STEP: 1

CAUSE:SB --TASK DESCRIPTION INADEQUACY
PRIMARY SYSTEM:PT --TEST/ CALIBRATION ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AF --PRESSURIZER (PWR)
COMPONENT:ISVZ--VALVE, ISOLATION/ SHUTOFF, UNKNOWN MATERIAL
VENDOR:E095--EDWARDS VALVES DIV
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:I --TESTING/ INSPECTION
EFFECT:AK --TRANSFER OPEN

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AF --PRESSURIZER (PWR)
COMPONENT:ORVZ--VALVE, RELIEF W/OPERATOR, UNKNOWN MATERIAL
VENDOR:T020--TARGET ROCK CORP.
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:HK --HIGH STATIC PRESSURE

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 3

SUBLINK:X
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AF --PRESSURIZER (PWR)
COMPONENT:ORVZ--VALVE, RELIEF W/OPERATOR, UNKNOWN MATERIAL
VENDOR:T020--TARGET ROCK CORP.
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:AK --TRANSFER OPEN

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 3

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AF --PRESSURIZER (PWR)

COMPONENT:PZR --PRESSURIZER
TIMING:A --ACTUAL IMMEDIATE
PERFORMANCE:T --TOTAL FAULT
DETECTION:F --OPERATIONAL ABNORMALITY
EFFECT:HL --LOW STATIC PRESSURE

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 5
CAUSE:VW --PROCEDURAL DEFICIENCY

INITIAL UNIT CONDITIONS: Z UNKNOWN/ NOT APPLICABLE
UNIT EFFECT: AC AUTOMATIC SCRAM

-----STEP: 7

THIS STEP IS DIRECTLY LINKED TO STEP 5
CAUSE:VW --PROCEDURAL DEFICIENCY

INITIAL UNIT CONDITIONS: Z UNKNOWN/ NOT APPLICABLE
UNIT EFFECT: EA AUTOMATIC ESF ACTUATION

-----STEP: 8

THIS STEP IS DIRECTLY LINKED TO STEP 7
SUBLINK:X
CAUSE:RE --SHUTDOWN/ ESF RESULTANT INFORMATION

INITIAL UNIT CONDITIONS: Z UNKNOWN/ NOT APPLICABLE
UNIT EFFECT: EA AUTOMATIC ESF ACTUATION

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 61 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
333 1990 012 1 9009050012 219453 04/04/90

DOCKET:333 FITZPATRICK TYPE:BWR
REGION: 1 NSSS:GE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: POWER AUTHORITY OF THE STATE OF NY
SYMBOL: PNY

COMMENTS

OTHER REPORTABILITY - VOLUNTARY. WATCH-LIST 933: GENERIC LETTER 89-13.

STEPS 2-5: MODEL NOS. B10-0114B-2T, W7-234B-2TY, W8-234B-2TY. STEP 13: COMP
MSC - VALVE DISC. STEPS 14,15: MODEL NO. 151AWE.

ABSTRACT

POWER LEVEL - 000%. DURING THE 1990 REFUEL OUTAGE, 61 CHECK VALVES IN

THE NORMAL (NSW) (KG) AND EMERGENCY SERVICE WATER (ESW) (BI) WERE
OPENED AND VISUALLY INSPECTED. THIRTY-SEVEN OF THE VALVES WERE
FROM

THE ASME SECTION XI IST PROGRAM AND 24 WERE FROM THE CHECK VALVE
PREVENTIVE MAINTENANCE (PM) PROGRAM. TWENTY IST AND 10 PM
PROGRAM

VALVES WERE INITIALLY DECLARED INOPERABLE DUE TO FAILING THE
VISUAL

INSPECTION CRITERIA. HOWEVER, THE IST CHECK VALVES WERE SHOWN TO
BE

OPERABLE BY ACTUAL FLOW TEST OR CALCULATION. OTHER EFFORTS
INCLUDED

INTERNAL INSPECTION OF 500' OF SMALL BORE PIPING AND 10
SAFETY-RELATED

COOLERS AND AIR HANDLING UNITS (AHUS). OF THESE COOLERS AND AHUS, 2

WERE FOUND TO HAVE 25% TUBE PLUGGING WITH SILT/SAND, BUT SHOWN
ABLE TO

REMOVE DESIGN BASIS HEAT LOAD. OF THE 500' OF PIPING, 200' WERE
FOUND 10%-30% RESTRICTED IN CROSS-SECTIONAL AREA, BUT A CALCULATION

DEMONSTRATED THAT FLOW CONTROL VALVES WERE HYDRAULICALLY
LIMITING. IN

ALL, THE ESW SYSTEM WAS CONSIDERED CAPABLE OF PERFORMING THE
DESIGN

SAFETY FUNCTION. THE AFFECTED VALVES, COOLERS, AND PIPING WERE
CLEANED OR REPLACED AS NECESSARY AND RETURNED TO SERVICE. THE
INTAKE

BAYS WERE ALSO CLEANED. PERIODIC FLUSHING AND PERFORMANCE TESTING

SHOULD PREVENT RECURRENCE. LERS 88-005, 88-009, AND 89-015 ARE
RELATED.

-----STEP: 1

CAUSE:SA --INTRINSIC HUMAN ERROR

PRIMARY SYSTEM:PD --DESIGN ACTIVITY

COMPONENT:PZ --UNKNOWN PERSONNEL

TIMING:M --ACTUAL PREEXISTING, UNDETECTED

PERFORMANCE:T --TOTAL FAULT

DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS

EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL

PRIMARY SYSTEM:KW --RAW SERVICE WATER

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL

VENDOR:V085--VELAN VALVE CORP.

QUANTITY:5 --

TRAIN:M --MULTIPLE

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:AR --FAILURE TO CLOSE

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION

PRIMARY SYSTEM:KW --RAW SERVICE WATER

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL

VENDOR:V085--VELAN VALVE CORP.

QUANTITY:5 --

TRAIN:M --MULTIPLE

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:AR --FAILURE TO CLOSE

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL

PRIMARY SYSTEM:CB --ESSENTIAL RAW COOLING/ SERVICE WATER

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL

VENDOR:V085--VELAN VALVE CORP.

QUANTITY:8 --

TRAIN:M --MULTIPLE

DIFFERENTIATOR:2 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:AL --FAILURE TO OPEN

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 4

SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:CB --ESSENTIAL RAW COOLING/ SERVICE WATER
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:V085--VELAN VALVE CORP.
QUANTITY:8 --
TRAIN:M --MULTIPLE
DIFFERENTIATOR:2 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AL --FAILURE TO OPEN

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:PIAZ--PIPE, LESS THAN 4 INCH DIAMETER, UNKNOWN
MATERIA
QUANTITY:2 --
TRAIN:M --MULTIPLE
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:HI --LOW FLOW

-----STEP: 7

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:CB --ESSENTIAL RAW COOLING/ SERVICE WATER
COMPONENT:PIAZ--PIPE, LESS THAN 4 INCH DIAMETER, UNKNOWN
MATERIA
QUANTITY:2 --
TRAIN:M --MULTIPLE
DIFFERENTIATOR:2 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:HI --LOW FLOW

-----STEP: 8

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:HA --REACTOR BUILDING HVAC
COMPONENT:CLR --COOLER

QUANTITY:4 --
TRAIN:M --MULTIPLE
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:HA --HIGH TEMPERATURE

-----STEP: 9

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PF --FABRICATION ACTIVITY
COMPONENT:PCP --CONTRACTOR PERSONNEL
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 10

THIS STEP IS DIRECTLY LINKED TO STEP 9

SUBLINK:A
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:P032--PACIFIC VALVES, INC.
DIFFERENTIATOR:3 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AR --FAILURE TO CLOSE

-----STEP: 11

THIS STEP IS DIRECTLY LINKED TO STEP 10

SUBLINK:X
CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:P032--PACIFIC VALVES, INC.
DIFFERENTIATOR:3 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AR --FAILURE TO CLOSE

-----STEP: 12

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PI --INSTALLATION ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL

DIFFERENTIATOR:3 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 13

THIS STEP IS DIRECTLY LINKED TO STEP 12
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:MSC --MISCELLANEOUS SUBCOMPONENT
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:K --MAINTENANCE
EFFECT:AB --MALADJUSTED/ MISPOSITIONED

-----STEP: 14

THIS STEP IS DIRECTLY LINKED TO STEP 13
SUBLINK:A
CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:C255--CHAPMAN VALVE & MFG
DIFFERENTIATOR:4 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AR --FAILURE TO CLOSE

-----STEP: 15

THIS STEP IS DIRECTLY LINKED TO STEP 14
SUBLINK:X
CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:KW --RAW SERVICE WATER
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:C255--CHAPMAN VALVE & MFG
DIFFERENTIATOR:4 --
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AR --FAILURE TO CLOSE

-----STEP: 16

THIS STEP IS DIRECTLY LINKED TO STEP 1
SUBLINK:A
CAUSE:PH --FOREIGN OBJECT/ MATERIAL
PRIMARY SYSTEM:CB --ESSENTIAL RAW COOLING/ SERVICE WATER

COMPONENT:PIBZ--PIPE, >4 AND <=6 INCH DIAMETER, UNKNOWN MATERIAL

QUANTITY:2 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:HI --LOW FLOW

-----STEP: 17

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL

PRIMARY SYSTEM:CB --ESSENTIAL RAW COOLING/ SERVICE WATER

COMPONENT:PIAZ--PIPE, LESS THAN 4 INCH DIAMETER, UNKNOWN MATERIAL

DIFFERENTIATOR:3 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:HI --LOW FLOW

-----STEP: 18

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL

PRIMARY SYSTEM:HA --REACTOR BUILDING HVAC

COMPONENT:AHU --AIR HANDLING/CONDITIONING UNIT (HEATING AND VENT

QUANTITY:2 --

TRAIN:M --MULTIPLE

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:HA --HIGH TEMPERATURE

-----STEP: 19

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:PH --FOREIGN OBJECT/ MATERIAL

PRIMARY SYSTEM:KW --RAW SERVICE WATER

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL

VENDOR:P032--PACIFIC VALVES, INC.

TRAIN:2 --

DIFFERENTIATOR:5 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:AR --FAILURE TO CLOSE

-----STEP: 20

THIS STEP IS DIRECTLY LINKED TO STEP 19

SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION

PRIMARY SYSTEM:KW --RAW SERVICE WATER

COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL

VENDOR:P032--PACIFIC VALVES, INC.

TRAIN:2 --

DIFFERENTIATOR:5 --

TIMING:P --POTENTIAL

PERFORMANCE:TR --TOTAL FAULT/REPAIR

DETECTION:I --TESTING/ INSPECTION

EFFECT:AR --FAILURE TO CLOSE

-----STEP: 21

THIS STEP IS DIRECTLY LINKED TO STEP 2 AND STEP 4

AND STEP 6 AND STEP 7 AND STEP 8 AND STEP 10

AND STEP 14 AND STEP 16

AND STEP 17 AND STEP 18 AND STEP 19

INITIAL UNIT CONDITIONS: H REFUELING

UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE

EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 62

LER SCSS DATA

12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE

333 1990 014 0 9005160149 218172 04/10/90

DOCKET:333 FITZPATRICK TYPE:BWR

REGION: 1 NSSS:GE

ARCHITECTURAL ENGINEER: SWXX

FACILITY OPERATOR: POWER AUTHORITY OF THE STATE OF NY

SYMBOL: PNY

COMMENTS

STEP 2: COMP MSC - SPRING.

ABSTRACT

POWER LEVEL - 000%. ON APRIL 10, 1990, A SPRING WAS DISCOVERED TO BE MISSING FROM A 2 INCH STOP AND SPRING ASSISTED PISTON CHECK VALVE

ABSTRACT

POWER LEVEL - 000%. ON APRIL 10, 1990, A SPRING WAS DISCOVERED TO BE MISSING FROM A 2 INCH STOP AND SPRING ASSISTED PISTON CHECK VALVE (23HPI-13). THE VALVE WAS OPENED DURING PERFORMANCE OF AN ASME SECTION XI IN-SERVICE TESTING PROGRAM DURING A PLANT REFUELING OUTAGE

WHICH STARTED MARCH 31, 1990. WITHOUT THE SPRING THE VALVE WAS INOPERABLE IN THE HORIZONTAL VALVE STEM ORIENTATION USED FOR THIS INSTALLATION. TECHNICAL SPECIFICATION TABLE 3.7-1 DESIGNATES THIS AS A CONTAINMENT ISOLATION VALVE. TABLE 4.7-2 EXEMPTS IT FROM TYPE "C" LOCAL LEAK RATE TESTING BECAUSE IT IS WATER SEALED. THE VALVE IS THE

LAST OF TWO CHECK VALVES IN THE DRAIN LINE FLOW PATH TO THE TORUS FROM

THE EXHAUST LINE OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) (BJ) TURBINE. THE ISOLATION FUNCTION WOULD HAVE BEEN PERFORMED BY THE

UPSTREAM CHECK VALVE ALTHOUGH IT IS NOT A DESIGNATED CONTAINMENT

ISOLATION VALVE. THE VALVE WAS OPENED FOR MAINTENANCE IN 1976 AND 1977 WITH NO RECORD OF SPRING REMOVAL. A CURRENT ENGINEERING REVIEW

SUPPORTS A CONCLUSION THAT TURBINE EXHAUST PRESSURE MAY NOT HAVE BEEN

SUFFICIENT UNDER SOME OPERATING CONDITIONS (DUE TO PRESSURE DROP THROUGH THE STEAM TRAP AND CHECK VALVES) TO OPEN THE CHECK VALVE WITH

THE SPRING INSTALLED. THE CHECK VALVE WILL BE ROTATED TO A STEM VERTICAL POSITION (WHICH DOES NOT REQUIRE A SPRING ASSIST TO CLOSE). THE SPRING WILL ALSO BE REMOVED FROM THE NEXT UPSTREAM CHECK VALVE.

-----STEP: 1

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PI --INSTALLATION ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:TA --OMISSION OF TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

SUBLINK:A

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION

SECONDARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:MSC --MISCELLANEOUS SUBCOMPONENT
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AA --MISSING

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PM --MAINTENANCE/ REPAIR ACTIVITY
COMPONENT:PUZ --UNKNOWN UTILITY PERSONNEL
DIFFERENTIATOR:2 --
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 3

SUBLINK:A
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:V085--VELAN VALVE CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AB --MALADJUSTED/ MISPOSITIONED

-----STEP: 5

THIS STEP IS DIRECTLY LINKED TO STEP 4

SUBLINK:X
CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:SD --REACTOR TORUS/ SUPPRESSION POOL (BWR)
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:V085--VELAN VALVE CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:AB --MALADJUSTED/ MISPOSITIONED

-----STEP: 6

THIS STEP IS DIRECTLY LINKED TO STEP 2 AND STEP 4

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:VOK --VALVE OPERATOR, PISTON
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:KB --FAILURE TO OPERATE

-----STEP: 7

THIS STEP IS DIRECTLY LINKED TO STEP 6

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:V085--VELAN VALVE CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:BI --BETWEEN SYSTEMS LEAKAGE, UNKNOWN RATE

-----STEP: 8

THIS STEP IS DIRECTLY LINKED TO STEP 7

SUBLINK:X

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:DB --CONTAINMENT ISOLATION
SECONDARY SYSTEM:SD --REACTOR TORUS/ SUPPRESSION POOL (BWR)
COMPONENT:CVZ --VALVE, CHECK, UNKNOWN MATERIAL
VENDOR:V085--VELAN VALVE CORP.
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:BI --BETWEEN SYSTEMS LEAKAGE, UNKNOWN RATE

-----STEP: 9

THIS STEP IS DIRECTLY LINKED TO STEP 7

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:PIZZ--PIPE, UNKNOWN DIAMETER, UNKNOWN MATERIAL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:HH --HIGH FLOW

-----STEP: 10

THIS STEP IS DIRECTLY LINKED TO STEP 9

CAUSE:NB --WATER SPRAY/ FLOOD/ CONDENSATION

PRIMARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:TRB --TURBINE
TIMING:P --POTENTIAL
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:KC --CESSATION OF OPERATION

-----STEP: 11

THIS STEP IS DIRECTLY LINKED TO STEP 10

CAUSE:RT --RESULTANT TOTAL SYSTEM FAULT
PRIMARY SYSTEM:BN --HIGH PRESSURE COOLANT INJECTION (BWR)
COMPONENT:XXX --ENTIRE SYSTEM
TIMING:P --POTENTIAL
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:HI --LOW FLOW

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

FORM 69 LER SCSS DATA 12-24-92

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
213 1991 023 1 9205190307 224793 11/10/91

DOCKET:213 CONNECTICUT YANKEE TYPE:PWR
REGION: 1 NSSS:WE
ARCHITECTURAL ENGINEER: SWXX
FACILITY OPERATOR: CONN. YANKEE ATOMIC POWER CO.
SYMBOL: CYA

COMMENTS

STEPS 2,3: INCONEL 600 SB-166 TUBES. STEP 4: MODEL NO. 27.

ABSTRACT

POWER LEVEL - 000%. SCHEDULED STEAM GENERATOR (SG) EDDY CURRENT TESTING (ECT) HAS BEEN CONDUCTED DURING THE 1991/1992 REFUELING OUTAGE
IN ACCORDANCE WITH ASME SECTION XI AND TECHNICAL SPECIFICATION 4.4.5.

ON NOVEMBER 9 AT APPROXIMATELY 1600 HOURS, WITH THE PLANT IN MODE 6

(REFUELING), IT WAS DETERMINED THAT ANALYZED ECT DATA PLACED SG NO. 2

INTO THE C-3 CATEGORY. A PROMPT REPORT OF THIS EVENT WAS MADE UNDER

10CFR50.72(B)(2)(I). SUBSEQUENTLY, ON NOVEMBER 10, 1991, ANALYZED ECT RESULTS FOR SG NO. 1 PLACED IT IN CATEGORY C-3, THUS REQUIRING EXPANSION OF THE ECT PROGRAM TO 100 PERCENT OF THE TUBES IN ALL FOURS

SG'S. THE MAJOR CAUSE OF THE TUBE DEFECTS WAS ATTRIBUTED TO PRIMARY

WATER STRESS CORROSION CRACKING AND, TO A LESSER EXTENT, PITTING. AS

A RESULT OF THE INSPECTIONS, A TOTAL OF 130 TUBES WERE PLUGGED. THIS SUPPLEMENTAL REPORT IS BEING ISSUED TO PROVIDE ADDITIONAL INFORMATION

ON THE IDENTIFIED CAUSES AND CORRECTIVE ACTION TAKEN. THIS EVENT IS

REPORTABLE PER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

-----STEP: 1

CAUSE:SZ --UNKNOWN HUMAN FACTOR CAUSE
PRIMARY SYSTEM:PZ --UNKNOWN ACTIVITY
COMPONENT:PZ --UNKNOWN PERSONNEL
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:T --TOTAL FAULT
DETECTION:R --REVIEW OF PROCEDURE/ TEST RESULTS
EFFECT:UA --COMMISSION OF UNDESIRE TASK, ANALYSIS, OR STEP

-----STEP: 2

THIS STEP IS DIRECTLY LINKED TO STEP 1

CAUSE:CA --MECHANICAL OVERLOAD
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:130 --
TRAIN:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 3

THIS STEP IS DIRECTLY LINKED TO STEP 2
SUBLINK:X

CAUSE:EF --CORROSION/ OXIDATION
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:TBG --TUBING
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:130 --
TRAIN:M --MULTIPLE
TIMING:M --ACTUAL PREEXISTING, UNDETECTED
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:DE --CRACK

-----STEP: 4

THIS STEP IS DIRECTLY LINKED TO STEP 2

CAUSE:RC --RESULTANT COMPONENT FAULT
PRIMARY SYSTEM:AH --STEAM GENERATOR (PWR)
COMPONENT:SG --STEAM GENERATOR
VENDOR:W120--WESTINGHOUSE ELECTRIC CORP.
QUANTITY:4 --
TRAIN:M --MULTIPLE
TIMING:P --POTENTIAL
PERFORMANCE:TR --TOTAL FAULT/REPAIR
DETECTION:I --TESTING/ INSPECTION
EFFECT:BD --WITHIN SYSTEM LEAKAGE, UNKNOWN RATE

INITIAL UNIT CONDITIONS: H REFUELING
UNIT EFFECT: XX NO SIGNIFICANT EFFECT

EFFECT ON ENVIRONMENT: N NO RELEASE
EFFECT ON PERSONNEL: N NO EXPOSURE

Appendix B

NPRDS Information

B-2

Appendix B

NPRDS Information

Nuclear Plant Reliability Data System Query Report

For: Peggy Cross-Prather
U.S. Nuclear Regulatory Commission

Report-Id: NPRS01AA
Job Number: 7590
Run Date: 03/01/93
Run Time: 15:51

Introduction:

The attached report was generated by your query of the NPRDS data base.
A summary of your query is listed below.

QUERY

You selected the following search condition(s):

Selected Safety Classs are Safety Class 1, Safety Class 2, and
Safety Class 3
Selected Detection is In-Service Inspection/Testing

DISPLAY AND SORT:

The following fields are displayed in the order you specified. The sort
sequence you chose is also indicated.

Sort Sequence -----	Field Name -----
	Number of Failures Component

Comp	Number Failure
-----	-----
ACCUMU	12
BLOWER	11
CKTBRK	9
CONROD	3
CRDRVE	49
ENGINE	20
FILTER	5
GENERA	6
HEATER	1
HTEXCH	205
IBISSW	30
ICNTRL	7
INDREC	4
INTCPM	3
IXMITR	24
MECFUN	6
MOTOR	15
PENETR	17
PIPE	99
PUMP	146
RECOMB	1
RELAY	3
SUPORT	336
TURBIN	7
VALVE	1755
VALVOP	239
VESSEL	25
=====	
	3038

1068 total 33 class 2 7 class 3 59 class 1

178 Total
out of 146, 107 total

<i>Component Code</i>	<i>Component Type</i>
ACCUMU	Accumulators
AIRDRY	Air/Gas Dryers
BATTRY	Batteries and Chargers
BLOWER	Blowers
CKTBRK	Circuit Closers/Interrupters
CONROD	Control Rods
CRDRVE	Control Rod Drive Mechanisms
DEMIN	Demineralizers
ENGINE	Engines, Internal Combustion
FILTER	Filters
GENERA	Generators
HEATER	Heaters, Electric
HTEXCH	Heat Exchangers
IBISSW	Instrumentation, Bistable/Switch
ICNTRL	Instrumentation, Controllers
INDREC	Instrumentation, Indicators/Recorders
INTCPM	Instrumentation, Integrator/Computation Module
IPWSUP	Instrumentation, Electric Power Supply
ISODEV	Instrumentation, Isolation Devices
IXMITR	Instrumentation, Transmitter/Primary Detector/Element

<i>Component Code</i>	<i>Component Type</i>
MECFUN	Mechanical Function Units
MOTOR	Motors
PENETR	Penetrations, Primary Containment
PIPE	Pipes, Fittings
PUMP	Pumps
RECOMB	Recombiners
RELAY	Relays
TRANSF	Transformers
TURBIN	Turbines
VALVE	Valves and Dampers
VALVOP	Valve Operators
VESSEL	Vessels, Pressure

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	CONNECTICUT YANKEE 1	WAD	SWS	WS-151-151 06/01/88	1
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The six-inch service water return pipe from the number 4 containment air recirculation fan had a thru wall leakage near pipe support PS-7. The hole was discovered after removing the insulation for inspection. The hole was a result of pitting on the outside diameter of the pipe. The exact cause of pitting was unknown. The hole was approximately one tenth of an inch. The pipe is scheduled to be repaired by welding a sleeve on the pipe surface in conjunction with weld overlay over pitting areas.

PIPE	DRESDEN 3	RBA	0300	3-0308-3 03/20/78	1
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During unit refueling outage, ultrasonic inspection of safe end-to-pipe weld on CRD return line revealed indications of cracking. The cause of the cracking is due to fatigue. The pipe was removed and replaced.

PIPE	FITZPATRICK 1	EEADAA	93	93-AEG-301-36A 01/15/91	1*
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During unit shutdown while performing inservice inspection pressure testing of 'A' emergency diesel generator air start system piping, air compressor 'A2' discharge check valve piping was found leaking air in excess of procedural limits (zero leakage). Though process fluid containment function was degraded, system was fully operable. Plant operation was not affected. Air was found to be leaking from a threaded pipe joint on the valve discharge piping. Actual cause of leakage was not determined. Check valve discharge piping was removed, pipe threads were cleaned and new sealant was applied. Piping was reinstalled and threaded joints were snoop tested and verified leak free per ISI procedural requirements.

PIPE	HOPE CREEK 1	CBA	BB	1-BB-013-VCA-28 02/19/91	1
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The plant was in mode 5, refueling outage 4, when the inservice test group reported that during penetrant examination of an 'A' loop reactor recirculation pipe weld linear indications were discovered that exceeded ASME Code allowance. There was no significant affect on the system or operation of the plant. The component was degraded and repair was required prior to plant startup. The defects/cracks were attributed to low ferrite content in the weld material during spool fabrication by the manufacturer. The low ferrite caused hot-short cracking. A deficiency report (DR) HMT-92-109, was issued and dispositioned. The system engineer notified the supplier and a metallurgical analysis of the weld sample was performed. The DR disposition included Code Job Package (CJP) H92-090. The linear indications were removed and prepared for rewelding, the reweld was blended, satisfactorily tested, and returned to service. (92-11-90); WO-920923144

* It appears an error was made in noting this as Safety Class 1.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/Date</u>	<u>Sfty Class</u>
PIPE	INDIAN POINT 3	WAD	SWS	LINE-12 05/01/88	1
Numerous pinhole leaks were found on piping of fan cooler unit service water discharge sample valve SWN-61-1 during preparation for system hydro test. The leak was not enough to effect the system or plant. The case of the failure was corrosion of 304 stainless steel pipe. The pipe was replaced. (17046)					
PIPE	MILLSTONE 1	CFA	SD	CCBJ-6 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified a crack adjacent to weld, but not thru wall. Pipe inside drywell on supply line from LPCI system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFA	SD	CUAJ-1 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified a crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to cleanup system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFA	SD	SCAJ-CU1 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified a crack adjacent to weld (weld-o-let), but not thru wall. Pipe inside drywell supply shutdown cooling system primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFA	SD	SCAJ-1 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified a crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to shutdown cooling system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFA	SD	SCAJ-2 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified a crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to shutdown cooling system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	MILLSTONE 1	CFB	IC	ICAJ-1 11/06/80	1
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During Class 1 & 2 ISI liquid penetrant examination on pipe, identified crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-10 11/06/80	1
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During Class 1 & 2 ISI liquid penetrant examination on pipe, identified crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-11 11/06/80	1
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During Class 1 & 2 ISI liquid penetrant examination on pipe, identified crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-14 11/06/80	1
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During Class 1 & 2 ISI liquid penetrant examination on pipe, identified crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-2 11/06/80	1
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During Class 1 & 2 ISI liquid penetrant examination on pipe, identified crack adjacent to weld, but not thru wall. Pipe inside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-3 11/06/80	1
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Same as 1 CAJ-2 above.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-5 11/06/80	1
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Same as 1 CAJ-2 above.

PIPE	MILLSTONE 1	CFB	IC	ICAJ-6 11/06/80	1
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Same as 1 CAJ-2 above.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
PIPE	MILLSTONE 1	CFB	IC	ICAJ-7 11/06/80	1
Same as 1 CAJ-2 above.					
PIPE	MILLSTONE 1	CFB	IC	ICAJ-8 11/06/80	1
Same as 1 CAJ-2 above.					
PIPE	MILLSTONE 1	CFB	IC	ICAJ-9 11/06/80	1
Same as 1 CAJ-2 above.					
PIPE	MILLSTONE 1	SDC		SCAJ-CU-1 11/06/80	1
During Class 1 & 2 ISI liquid penetrant examination on pipe, identified cracks on both sides of welds, but not thru wall. Pipe inside drywell on supply to cleanup system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFA		SDCC-146 02/09/81	1
A flaw indication in the piping on LPCI B loop was found between penetration and inboard isolation check valve. Weld degradation is believed to be intergranular stress corrosion cracking. However, some transgranular cracking may be present. Replaced piping section from penetration to inboard isolation check valve (304 SS).					
PIPE	MILLSTONE 1	SFA	LPCI	C.S.SPGR.D 09/25/82	1
A crack has been identified on one core spray sparger junction box weld. Believed to be high weld residual stress. Stress corrosion cracking may have played a part. Installed sparger hold clamp. (These clamps have been installed at other plants for same problem.)					
PIPE	MILLSTONE 1	SDC		CUAJ-11 04/27/84	1
A crack indication was detected by ultrasonic testing of an 8 inch cleanup system line. The indication measured. 0.08 in. deep by 0.75 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.02 carbon) schedule 80 pipe.					
PIPE	MILLSTONE 1	SDC		CUAJ-15 04/27/84	1
A crack indication was detected by ultrasonic testing of a 8 inch cleanup system line. The indication measured 0.34 in. deep by 6 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.02 carbon) schedule 80 pipe.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	MILLSTONE 1	SDC		CUAJ-19 04/27/84	1
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A crack indication was detected by ultrasonic testing of a 8 in. cleanup system line. The indication measured 0.30 in. deep by 10 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.02 carbon) schedule 80 pipe.

PIPE	MILLSTONE 1	SDC		CUAJ-19A 04/27/84	1
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A crack indication was detected by ultrasonic testing of a 8 in. cleanup system line. The indication measured 0.39 in. deep by 360 degrees in length. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.02 carbon) schedule 80 pipe.

PIPE	MILLSTONE 1	SDC		CUAJ-22 04/27/84	1
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A crack indication was detected by ultrasonic testing of a 8 in. cleanup system line. The indication measured 0.08 in. deep by 1.75 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.035 carbon) schedule 160 pipe (90 degree elbow).

PIPE	MILLSTONE 1	SDC		CUAJ-24 04/27/84	1
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A crack indication was detected by ultrasonic testing of a 8 in. cleanup system line. The indication measured 0.11 in. deep by 2.5 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.035 carbon) schedule 160 pipe (90 degree elbow).

PIPE	MILLSTONE 1	SDC		CUAJ-25 04/27/84	1
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A crack indication was detected by liquid penetrant testing of an 8 inch cleanup system line. The indication measured as 5 linear indications. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.035 carbon) schedule 160 pipe (90 Degrees elbow).

PIPE	MILLSTONE 1	SDC		CUAJ-9 04/27/84	1
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A crack indication was detected by ultrasonic testing of an 8 inch cleanup system line. The indication measured 0.11 in. deep by 2 in. long. Intergranular stress corrosion cracking is suspected; within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss (with < 0.02 carbon) schedule 80 pipe.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
PIPE	MILLSTONE 1	SFA	LPCI	CSBC-G-1 04/27/84	1
A crack indication was detected by ultrasonic testing of a 10 inch core spray line. The indication measured 0.1 in. deep by 360 degrees intermittent. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and was replaced with type 304 ss with < 0.02 carbon-schedule 80 pipe.					
PIPE	MILLSTONE 1	SFA	LPCI	CSBJ-12 04/27/84	1
A crack indication was detected by ultrasonic testing of a 10 in. core spray line. The indication measured 0.25 in. deep by 6 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and was replaced with type 304 ss with < 0.02 carbon-schedule 80 pipe.					
PIPE	MILLSTONE 1	CBA	301	RRAJ-4 05/07/84	1
A crack indication was detected by ultrasonic testing of a 12 inch reactor recirculation riser pipe. The indication measured 0.20 inch deep by 4 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. A licensee event report was submitted.					
PIPE	MILLSTONE 1	CBA	301	RRBJ-1 05/07/84	1
A crack indication was detected by ultrasonic testing of a 12 inch reactor recirculation riser pipe. The indication measured 0.03 inch deep by 1.5 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. Also induction heat stress improvement process was completed on the affected area. A licensee event report was submitted.					
PIPE	MILLSTONE 1	CBA	301	RRBJ-2 05/07/84	1
A crack indication was detected by ultrasonic testing (also a visual leak) of a 12 inch reactor recirculation riser pipe. The indication measured thru-wall by 1/4 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. Also induction heat stress improvement process was completed on the affected area. A licensee event report was submitted.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	MILLSTONE 1	CBA	301	RRCJ-3 05/07/84	1
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A crack indication was detected by ultrasonic testing of a 12 inch reactor recirculation riser pipe. The indication measured 0.10 inch deep by 5 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. Also induction heat stress improvement process was completed on the affected area. A licensee event report was submitted.

PIPE	MILLSTONE 1	CBA	301	RREJ-3 05/07/84	1
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A crack indication was detected by ultrasonic testing of a 12 inch reactor recirculation riser pipe. The indication measured 0.32 inch deep by 5 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. A licensee event report was submitted.

PIPE	MILLSTONE 1	CBA	301	RRFJ-1 05/07/84	1
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A crack indication was detected by ultrasonic testing of a 12 inch reactor recirculation riser pipe. The indication measured 0.13 inch deep by 12 inch long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process. A licensee event report was submitted.

PIPE	MILLSTONE 1	CBA	301	JPAF-2-SE 11/22/85	1
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During inservice inspection, found ultrasonic indications of jet pump instrument assembly 4 inch safe end to 4 inch x 8 inch eccentric reducer. 0.042 inch deep x 0.5 inch long. Intergranular stress corrosion cracking. Weld cracks were repaired by weld overlay technique and pressure tested satisfactorily. A licensee event report was submitted.

PIPE	MILLSTONE 1	CBA	301	JPAJ-2 11/22/85	1
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During inservice inspection, found ultrasonic indications on jet pump instrument assembly 12 inch eccentric reducer to 12 inch pipe. Flaws 1 and 2 are 0.1 inch x 4.66 inch. Intergranular stress corrosion cracking. Weld cracks were repaired by weld overlay technique and pressure tested satisfactorily.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
PIPE	MILLSTONE 1	CBA	301	JPBF-2-SE 11/22/85	1
During inservice inspection, found ultrasonic test indications (IGSCC) on jet pump instrument assembly 4 inch safe end to 4 inch x 8 inch eccentric reducer. Flaw 1 - 0.09 inch x 0.5 inch. Flaw 2 - 0.06 inch x 0.25 inch. Intergranular stress corrosion cracking. Weld cracks were repaired by weld overlay technique and pressure tested satisfactorily.					
PIPE	MILLSTONE 1	CBA	301	JPBJ-2 11/22/85	1
During inservice inspection, found ultrasonic test indications on jet pump instrument assembly 12 inch eccentric reducer to 12 inch pipe. 0.172 inch deep x 360 deg. Intergranular stress corrosion cracking. Unit in refueling. Weld cracks were repaired by weld overlay technique and pressure tested satisfactorily. A licensee event report was submitted.					
PIPE	MILLSTONE 1	CBA	301	JPBJ-3 11/22/85	1
During performance on inservice inspection, found ultrasonic test indications of jet pump instrument assembly 12 inch pipe to 12 inch seal. 0.344 inch deep x 360 deg. Intergranular stress corrosion cracking. Unit in refueling. Weld cracks were repaired by weld overlay technique and pressure tested satisfactorily. A licensee event report was submitted.					
PIPE	MILLSTONE 1	CBA	301	RMBJ-1 06/26/87	1
During refueling, a crack indication was found on a pipe to cap weld. The indication was found unacceptable and needed repair. The weld was last inspected during the refueling in 1984. The cause is believed to be intergranular stress corrosion cracking. The weld indication in question was weld overlayed with ER308l which has immunity to igsc. A system hydrostatic test will be performed. A licensee event report was submitted.					
PIPE	MONTICELLO 1	SFA	B.3.1	TW7-8 05/01/81	1
During outage, while performing annual ISI inspection on core spray loop B, several non-through-wall crack indications were discovered. Apparent cause of cracks was intergranular stress corrosion cracking. Replaced part of safe end, safe end reducer, and all stainless steel piping up to the first manual isolation valve on both core spray loops.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/Date</u>	<u>Sfty Class</u>
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PIPE	MONTICELLO 1	CBA	B.1.4	REW-14-12 09/28/82	1
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During ISI inspection while in cold shutdown for refuel, IGSCC indications were found on recirc. system piping safe ends and elbows. Cause was due to intergranular stress corrosion cracking in sensitized 304 stainless steel base material. Cracks were weld repaired using a weld overlay process and E308L weld filler material.

PIPE	NINE MILE POINT 1	SFA	80,80.1	CSLOOP-11 05/07/84	1
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During ultrasonic testing of the core spray system, a crack in #11 loop pipe was discovered. The cracks were caused by chlorine induced stress corrosion. The cracks were ground out and the cavity weld repaired and ground smooth; retesting showed satisfactory results.

PIPE	PEACH BOTTOM 2	CBA	02	N8-A 06/07/84	1
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Ultrasonic examination revealed a through-wall crack in the 'A' jet pump instrumentation penetration reducer to safe end weld. Analysis indicates a pattern typical of intergranular stress corrosion cracking. A modified reducer design will be installed during current primary system pipe replacement outage.

PIPE	PEACH BOTTOM 3	CBA	02	N8-A 06/10/84	1
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Ultrasonic examination revealed crack like indications in the 'A' jet pump instrumentation penetration reducer to safe end weld. Analysis indicates a pattern typical of intergranular stress corrosion cracking. Full structural weld overlay repairs were performed prior to returning the unit to service. Moisture detection devices were installed on the 'A' penetrations.

PIPE	PEACH BOTTOM 3	CBA	02	N8-B 06/10/84	1
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Ultrasonic examination revealed crack like indications in the 'B' jet pump instrumentation penetration reducer to safe end weld. Analysis indicates a pattern typical of intergranular stress corrosion cracking. Full structural weld overlay repairs were performed prior to returning the unit to service. Moisture detection devices were installed on the 'B' penetrations.

PIPE	SALEM 1	WAD	028	1-SW-72 07/04/91	1
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Unit 1 at 100 percent power. Maintenance personnel were performing a inspection on number 16 service water strainer discharge piping (1-SW-72), when they discovered a through the wall leak, and pitting of the inner wall. No train, or plant effect. This was a small hole in piping. Maintenance personnel have determined that failure was due to normal wear out of piping, agitated by the corrosive, erosive properties of the service water system. The through the wall leak was welded, and pitting also. Pipe was returned to service with no leaks observed. This report was sent to our system engineers for further evaluation. (901001147) TJW

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>S'ty Class</u>
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PIPE	SALEM 1	HHF	012-263	RD-1S-FWR-10 01/01/92	1
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Unit 2 at 100 percent power. Maintenance personnel were performing a in-service inspection on number 12 steam generator feed pump recirculation piping (RD-1S-FWR-10), and it was found to be below acceptable minimum wall thickness. This was determined to be a reportable failure by our system engineer. No train, or plant effect. Maintenance personnel suspect failure cause to be from normal wear of piping wall, agitated by corrosive, erosive properties of the feedwater system. Maintenance personnel replaced the piping in kind, and performed other preventive maintenance activities on the piping system. This report was sent to our system engineer for further review. (920613074) TJW

PIPE	SALEM 2	WAD	028	2-SW-P-126 07/04/91	1
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Unit two off line for feedwater piping evaluations. Maintenance personnel were performing a hydrostatic test on the service water return header piping (2-SW-P-126) for the chiller condensers, and a through the wall leak was discovered. This was a small leak, and did not effect the service water system, or the performance of the chiller condensers. No train, or plant effect. Maintenance personnel suspect failure cause to be from normal wear out of the piping wall, agitated by the corrosive, erosive properties of the service water system. Maintenance personnel replaced the piping in kind. No leaks were observed when the service water flow was returned. This report was sent to our system engineer for further review. (920707159) TJW

PIPE	SALEM 2	HHF	012-263	2SGF-113R1 06/16/92	1
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Unit two off line for feedwater piping evaluation. Our system engineers were performing a scheduled examination of the feedwater piping, and they discovered that the piping wall of the eccentric reducers down stream of number 23 steam generator inlet valve were thinning. No train, or plant effect. The piping was still able to perform it's function of fluid containment. This is considered incipient. Maintenance personnel suspect wall thinning is due to normal wear of piping wall, agitated by aging, corrosion, and erosion. Maintenance personnel replaced the eccentric reducers in kind. This report was sent our system engineer for further review. (920622124) TJW

PIPE	SALEM 2	HHF	012-263	2SGF-113R2 06/16/92	1
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Unit two off line for feedwater piping evaluation. Our system engineers were performing a scheduled examination of the feedwater piping, and they discovered that the piping wall of the eccentric reducers down stream of number 24 steam generator inlet valve were thinning. No train, or plant effect. The piping was still able to perform it's function of fluid containment. This is considered incipient. Maintenance personnel suspect wall thinning is due to normal wear of piping wall, agitated by aging, corrosion, and erosion. Maintenance personnel replaced the eccentric reducers in kind. This report was sent our system engineer for further review. (920622124) TJW

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	SALEM 2	WAD	028	2-SW-P-112 06/16/92	1
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Unit two off line for feedwater piping evaluations. Maintenance personnel were performing a hydrostatic test on the service water return header piping (2-SW-P-112) for the chiller condensers, and a through the wall leak was discovered. This was a small leak, and did not effect the service water system, or the performance of the chiller condensers. No train, or plant effect. Maintenance personnel suspect failure cause to be from normal wear out of the piping wall, agitated by the corrosive, erosive properties of the service water system. Maintenance personnel replaced the piping in kind. No leaks were observed when the service water flow was returned. This report was sent to our system engineer for further review. (920705090) TJW

PIPE	SALEM 2	WAD	028	2-SW-P-113 06/16/92	1
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Unit two off line for feedwater piping evaluations. Maintenance personnel were performing a hydrostatic test on the service water return header piping elbow (2-SW-P-112) for the chiller condensers, and a through the wall leak was discovered. This was a small leak, and did not effect the service water system, or the performance of the chiller condensers. No train, or plant effect. Maintenance personnel suspect failure cause to be from normal wear out of the piping wall, agitated by the corrosive, erosive properties of the service water system. Maintenance personnel replaced the piping elbow in kind. No leaks were observed when the service water flow was returned. This report was sent to our system engineer for further review. (920705090) TJW

PIPE	BRUNSWICK 2	SFC	2E41	2-E41-22-1-604 06/04/89	2
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With unit 2 in cold shutdown, maintenance personnel performing ISI testing on the high pressure coolant injection steam supply drain pot drain valve (2-E41-22-1-604) found the pipe had degraded to a wall thickness of 0.219 in. The required wall thickness for this pipe is 0.25 inch as per ASME Section III. The pipe is required to maintain system integrity. The discovery presented no problems to the system or plant since no actual leakage was found but engineering required the pipe to be replaced prior to unit 2 restart. The root cause of the wall thinning was due to normal erosion by water flow. Maintenance replaced the pipe (2-E41-22-1-604) and fittings with identical spares. Trk no 93-043 (bjb 90-aexk1)

PIPE	COMANCHE PEAK 1	SCJ	CT	BRP-CT-1-SB-004 05/23/91	2
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With plant at 0 percent power and the system and train -GSK-01 in test, the containment spray pump 1-03 discharge flow element orifice flange was found to be leaking externally by quality control inspector during Code leak inspection as evidenced by boron crystals on the flange. As the leak was only a seep, there was no effect on the system, train or plant. The orifice remained in service about 5 months until the first refueling outage. The cause of the failure was determined to be loose flange fasteners and/or misalignment of the 2 flanges due to previous maintenance activities. Disassembled flange, cleaned and inspected gasket surfaces, aligned flanges, replaced gaskets and torqued fasteners. Post work test showed the leak had stopped.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/Date</u>	<u>Sfty Class</u>
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PIPE	COMANCHE PEAK 1	SCJ	CT	BRP-CT-1-SB-017 05/23/91	2
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With plant at 0 percent power and system train in test -GSK-1, the containment spray pump 1-01 discharge flow element orifice flange was found to be leaking externally by quality control inspector during Code leak inspection as evidenced by boron crystals on the flange. As the leak was only a seep, there was no effect on the system train or plant. The orifice remained in service about 5 months until the first refueling outage. The cause of the failure was undetermined, but was believed to be loose flange fasteners and/or misalignment of the 2 flanges. Disassembled flange, cleaned and inspected gasket surfaces, aligned flanges, replaced gaskets and torqued fasteners. Post work test showed the leak to be stopped.

PIPE	HATCH 1	SCADIL	T48	T48-2CPI-18-PID 09/30/84	2
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With the unit in cold shutdown for refueling -6, in-service inspection was being performed on the nitrogen inerting and purge line to the drywell. A linear, through-wall crack approximately 2 3/4 inches long was discovered in weld 1T48-2CPI-18-PID-6. The crack was caused by thermally induced stress in the area of the weld. The stress was induced by nitrogen impinging the weld area and cooling it below the nil ductility transition temperature. The crack was ground-out and repaired. The weld was then inspected via magnetic particle method; no indication was found. The setpoint for low temperature alarm for the nitrogen inerting system will be raised from 0 degree F to 10 degree F and the system procedures will be revised to include instructions to discontinue nitrogen flow upon low temperature alarm.

PIPE	INDIAN POINT 3	SFK	SIS	FE-927 01/10/84	2
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A flange leak was discovered for SI flow orifice FE-927 during an inservice inspection surveillance test. The studs at the flanges were loose and corroded. The studs were replaced, retorqued, and retested.

PIPE	MAINE YANKEE 1	SCD	CSS	3"-CS-24-302R3 10/22/85	2
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During a refueling shutdown and radiographic inspection of the recirculation line for containment spray pump P-61A, line number 3 in.-CS-24-302R3 was found to have a defective weld at a 90 degree elbow. Dr 5317-85 The defect was possibly present since construction. Replaced the elbow section of piping.

PIPE	MILLSTONE 1	CHA	CND/FDW FW-SPARGERS	11/01/76	2
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Cracks in feed water spargers discovered by dye penetrant exam as part of in-service inspection program. Suspect stress-fatigue caused by improper design; replaced spargers.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	MILLSTONE 1	CFB	IC	ICAC-F-16A 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified a crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to isolation condenser system. Primary cause for failure is intergranular stress corrosion cracking. However, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-17 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified a crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-18 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified a crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-19 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified a crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-25A 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-26 11/06/80	2
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During Class 2 ISI, liquid penetrant examination on pipe identified crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-27 11/06/80	2
During Class 2 ISI, liquid penetrant examination on pipe identified crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-28 11/06/80	2
During Class 2 ISI, liquid penetrant examination on pipe identified crack adjacent to weld, but not thru wall. Pipe outside drywell on supply to Isol. Cond. Sys. Primary cause for failure is intergranular stress corrosion cracking; however, substantial transgranular cracking was evident. Replaced pipe section with low carbon stainless steel pipe. Tested satisfactorily.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-22 04/19/84	2
A crack indication was detected by ultrasonic testing of a 12 in. isolation condenser line. The indication measured 0.11 in. deep by 2 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and was replaced with type 304 ss with < 0.02 carbon-schedule 80 pipe.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-25 04/19/84	2
A crack indication was detected by ultrasonic testing of a 12 in. isolation condenser line. The indication measured 0.14 in. deep by 2.75 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and was replaced with type 304 ss with < 0.02 carbon-schedule 80 pipe.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-3 04/19/84	2
A crack indication was detected by ultrasonic testing of a 16 in. isolation condenser line. The indication measured 0.18 in. deep by 4 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was repaired by weld overlay process.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-4 04/19/84	2
A crack indication was detected by ultrasonic testing of a 16 in. isolation condenser line. The indication measured 0.36 in. deep by 7 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and replaced with type 316l ss-schedule 120 pipe. Note: this is an elbow 90 degrees.					
PIPE	MILLSTONE 1	CFB	IC	ICAC-F-5 04/19/84	2
A crack indication was detected by liquid penetrant testing of a 16 in. isolation condenser line. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and replaced with type 316l ss-schedule 120 pipe.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	MILLSTONE 1	CFB	IC	ICAC-F-6 04/19/84	2
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A crack indication was detected by ultrasonic testing of a 16 in. isolation condenser line. The indication measured 0.15 in. deep by 9 in. long. Intergranular stress corrosion cracking is suspected within the heat affected zone. The affected area of pipe was removed and replaced with type 304 ss with < 0.02 carbon-schedule 80 pipe.

PIPE	MILLSTONE 1	CFB	IC	ICAC-F-13 10/31/85	2
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During inservice inspection on isolation condenser Class 2 steam supply line, a 12 in. pipe weld joint was found to have an unacceptable crack indication by ultrasonic testing. Believed to be intergranular stress corrosion cracking. The flaw ~ 0.22 in. deep x 2.5 in. long. Two other areas within this weld area, but not as deep into pipe wall. The defective weld was weld overlayed in accordance with ASME Code Section XI, 1980 edition. The pipe will be hydrostatically tested before return to service.

PIPE	MILLSTONE 1	CFB	IC	ICBC-F-14 07/02/87	2
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An ultrasonic test identified an unacceptable inclusion in the base metal of the isolation condenser return line. This 8 inch diameter pipe extends axially for approximately 26 inches. Intergranular stress corrosion cracking of the isolation condenser return line pipe. The pipe section was replaced using a material that is more resistant to intergranular stress corrosion cracking. A system hydrostatic test was performed.

PIPE	OYSTER CREEK 1	SFA	212	NZ-3-38 04/10/89	2
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Mode switch in startup. While performing the 1000 lb inspection a pinhole was discovered at core spray system weld #NZ-3-38. Leak was at toe of weld to base metal. Weld configuration is pipe to elbow. 8 in. stainless steel butt weld. Test results revealed that the pinhole leak had characteristics of intergranular stress corrosion cracking. Performed complete circumferential weld overlay. Performed ultrasonic testing and a visual at 1000 lb. Startup was delayed until 5/11/89 because of this.

PIPE	QUAD CITIES 2	CBA	0220	2-201M-12A 05/30/84	2
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Unit 2 was in the refuel mode. On 4-1-85 a visual inspection revealed a recirculation weld area, pipe 2-201M-12A, with water seeping from a small crack. The unit was shutdown for refueling and maintenance making the safety consequences of this event minimal. The cause of the crack indication is postulated to be intergranular stress corrosion cracking. Materials used in original recirculation piping & fittings are regular grade type 304 stainless steel which is known to be susceptible to stress corrosion cracking based on plant operating history. Pipe was installed by Dravo Corp. The pipe was repaired by a weld overlay/flaw removal. All analyses were performed to the guidelines specified in the ASME Code, Section XI. Prior to reactor startup, the recirculation system was hydrostatically tested in conjunction with the reactor vessel hydrostatic test at 1.1 times the system nominal operating pressure.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	QUAD CITIES 2	CBA	0220	2-201K-12A 12/01/85	2
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On 11-5-86 while conducting a visual inspection of prepared weld surface prior to ultrasonic examination, a small pinhole leak was found on recirculation weld 02K-S3. Ultrasonic examination confirmed the leak to be an axial oriented crack. Safety implications were minimal since unit 2 was in a refueling outage. Utility component ID-pipe 2-201K-12A. The exact cause of the crack indication has not been determined but it is postulated that intergranular stress corrosion cracking is the mode of failure. 02K-S3 was repaired with a 'full structural' design overlay. Overlay was surface finished to permit application of EPRI technique for overlay ultrasonic examination which examines the overlay metal and the upper 25% of the original piping material. A hydrostatic test of 1110 psig was conducted prior to unit startup and no leaks were found.

PIPE	SAN ONOFRE 1	PCG		CS 2002-3-2501 10/23/78	2
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With the unit refueling, engineering was performing inservice inspection testing of the charging pump discharge line and noted water seeping into an adjacent piping tunnel from the sand filled pipe trench where the discharge line was located. The sand was removed from the trench and the leaks identified by liquid penetrant and ultrasonic testing. Caused by through-wall cracking as a result of chloride stress corrosion. The piping was buried in sand with detectable chlorides which ranged from 70 to 1000 parts per million. The affected areas of the pipe were replaced with in-kind pipe of A312 type 316 schedule 160 stainless steel. The external environment (sand in the pipe trench) was not replaced. The trench was left clear and covered with concrete plugs. The piping was tested and returned to service satisfactorily.

PIPE	SAN ONOFRE 1	PCG	CS	2014-3-151R 10/23/78	2
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With the unit refueling, engineering was performing inservice inspection testing on the reactor coolant pump seal water return line and noted water seeping into an adjacent piping tunnel from the sand filled pipe tunnel where the seal water line was located. The sand was removed and the leak identified by liquid penetrant and ultrasonic testing. The cause: through-wall cracking as a result of chloride stress corrosion. Piping was buried in sand with detectable chlorides which ranged from 70 to 1000 ppm. The seal water line originally consisted of A312 type 304 schedule 10 stainless steel. The affected areas of the pipe was replaced with A312 type 316 schedule 40 stainless steel. The external environment (sand in the pipe trench) was not replaced. The pipe trench was covered with concrete vault. The piping was tested satisfactorily and returned to service.

PIPE	SAN ONOFRE 1	HHC	FC	326-10-EG 11/25/78	2
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With the unit at 100% power, operations stated that the channel 'C' feedwater flow indicator exhibited a sudden increase in flow indication resulting in loss of one channel of the steam flow/feed flow mismatch trip. Investigation using radiographic examination revealed that the feedwater piping flow straightener had become dislodged from its normal position and had moved downstream where it was stopped by the feedwater flow orifice plate. Corrective action was to initiate continuous operator surveillance. The flow straightener will be repaired or replaced at the next available outage.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	SAN ONOFRE 1	SCJ	CRS	CONT-SPRAY 01/04/89	2
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Unit was refueling with the containment spray system in test. The air flow test indicated 7 nozzles were clogged with a foreign material. The system operation was degraded, but plant operation was not affected due to refueling mode. Investigations determined the blockage of the CSS piping and nozzles was due to material comprised of sodium silicate coating which had been applied in 1977. The coating had deteriorated and separated from the CSS piping walls over time. The coating had been applied to reduce and stop corrosion. MO90112135, 90120670, NCR90070015. All affected CSS piping and nozzles were cleaned to remove the coating to the extent possible. A combination of brushing and chemical cleaning was used. An air flow test was performed satisfactorily, and the system was returned to service.

PIPE	TURKEY POINT 3	CFF	RHR	3-887/10-IN-SI- 02/21/86	2
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Unit 3 was defueled during ten year hydrostatic testing of the residual heat removal system. Ten pinhole-sized leaks were found during visual inspection of a ten inch piping elbow in the recirculation line 3-887 10-IN-SI-151R-elbow from the heat exchangers to the pumps. The leaks degraded the fluid containment function of the elbow and resulted in failure of the hydrostatic test. The system and unit were not affected. Cause of the pin hole leaks is not known. Suspect at this time a defect in the type 304 stainless steel material. Root cause is pending analysis. The defective piping elbow was cut out of and removed from the system and saved for root cause analysis. The elbow (90 degree, ten inch schedule 40, type 304 stainless steel) was replaced like for like and welded into the system. The system was released for testing. *WA910221092541, non-conformance report NCR-N91-0220, hydrostatic test number 03-6205-H-01.

PIPE	TURKEY POINT 3	CFF	RHR	3T229C/3C-ACCUM 01/29/91	2
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Unit 3 was defueled during inservice inspection hydrostatic testing of the '3C' accumulator. Released leakage from the 'nitrogen vent and supply line' to the accumulator was noted. The leakage was located on the piping between the accumulator connection nozzle and the first isolation valve (3-883H) and appeared as small beads that reappeared after wiping. The piping leakage degraded the '3C' accumulator train. The unit was not affected. Cause of the pipe leakage was not known. Failure root cause investigation is in progress. Suspect abnormal stress, normal wear or aging/cyclic fatigue at this time. The leaking section of pipe between two 'elbow' fittings was removed and replaced (one inch, schedule 40, A312 type 304 seamless). The new piece was welded in. The pipe and train were released to operations for testing. Re: non-conformance report N91-0084 *wa910129092507

PIPE	TURKEY POINT 4	SDB	CIS	SL-2505B-4/PEN- 04/01/85	2
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While unit 4 was at full power, a small leak was observed at a 'tee' fitting for valve 4-991, 'pressurizer steam space sample line test connection'. The 'tee' is located just outside containment penetration eight for the pressurizer steam space sampling line. The leaking fitting degraded the containment isolation function of the piping and one train of the containment isolation system; however, the sampling system and the unit were not significantly affected. The leak was discovered during an in-service inspection of the piping. The cause of failure is unknown. Suspect vibration as the cause. The 'tee' connector was replaced and the leakage was stopped. *6978-64

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	CALLAWAY 1	WBD	EG	BBFE0020 08/16/91	3
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With the unit in a refueling outage and the component cooling water system isolated for a hydrostatic test, inspection personnel discovered through in-service testing that the flow element on the cold water outlet for the reactor coolant pump 'd' thermal barrier cooling coil had a gasket leak. Due to the minor nature of the leak there was no significant affect on system or plant operations. Cause of the failure is unknown. Probable cause is normal gasket wear. Removed the flow element, cleaned all gasket seating surfaces and installed new gaskets. The flow element was reinstalled, appropriate retests performed and returned to service. (W513699)

PIPE	MAINE YANKEE 1	HBG	FWS	12-S1E-2-301 01/29/92	3
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With the plant in an unscheduled shutdown it was discovered that the pup piece for the E-11B first point feedwater heater of the main feedwater system had eroded to the point that it was in need of replacement. This was discovered during ultrasonic thickness testing. The pup piece is a section of pipe that makes up the difference between the feedwater heater inlet nozzle and the inlet steam pipe. The E-11b is one of two trains of heaters, only one of which is needed for plant operations. Although this pipe did not fail, it was degraded to the point that further erosion would result in its inability to withstand the system pressure. The system was still operable. There was no effect upon the plant operations. Cause: expected age, wear, and erosion of associated feedwater piping are the causes of this failure. Action: it was decided to take advantage of the unscheduled shutdown to perform this repair. The old pup piece was removed and a new one was welded into place. (920472400)

PIPE	MILLSTONE 1	WAA		SW-39 03/21/89	3
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There was excessive leakage in the service water system header down stream of the main service water strainer. The leak was identified while pressurizing the header for the 10 year in-service inspection hydrostatic testing. There was no significant effect on the plant, but this train of the service water system was degraded. The leakage was discovered in the weld area of the 20 inch pipe SW-39 and was caused by erosion. The washout area of the weld was ground out and rewelded. No detectable leakage resulted when tested.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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PIPE	NORTH ANNA 2	WAD		2-SW-PIPE-444 06/19/89	3
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With the reactor at power during the performance of an inservice inspection hydrostatic test, the service water supply pipe to the containment penetration coolers developed a pin hole leak. The system was not affected as this system is a backup means of cooling only. There was no significant effect on overall plant operations. This failure was caused by internal corrosion of the pipe. Cut out and replaced the affected section of pipe. The component was tested satisfactorily and returned to service. (89-504)

PIPE	OYSTER CREEK 1	CFASCK 241		10-SW-2 07/10/85	3
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On July 22, 1985 the plant was brought to a cold shutdown condition from 41.3% power due to both containment spray systems 1 & 2 being declared out of service. The cause of failure is attributed to accumulation of emergency service water (ESW) pipe coal tar lining material fragments blocking the inlet tubesheets of containment spray heat exchangers CSHX 1-3 & 1-4. The coating failure is attributed to repeated thermal cycles of the outdoor exposed portion of ESW piping while drained during RO-10 outage. The ESW pipe downstream of pumps 52c & 52d was inspected and hydrolazed with high pressure water to remove the coating. Approximately 50 feet of piping was cleaned. The heat exchangers 1-3 & 1-4 were also opened and cleaned.

PIPE	OYSTER CREEK 1	WAA	531	10-SW-2 07/21/85	3
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On July 22, 1985 the plant was brought to a cold shutdown condition from 41.3% reactor power due to both containment spray systems 1 and 2 being declared out of service. The cause of failure was attributed to accumulation of emergency service water (ESW) pipe coal tar lining material fragments blocking the inlet tubesheets of containment spray heat exchangers CSHX1-1 and 1-2. The coating failure is attributed to repeated thermal cycles of the outdoor exposed portion of ESW piping while drained during the RO-10 outage. The ESW pipe downstream of pumps 52A & 52B was inspected and hydrolazed with high pressure water to remove the coating. Approximately 50 feet of piping was cleaned. The heat exchangers 1-1 & 1-2 were also opened and cleaned.

PIPE	SEABROOK 1	WAD	SW	1SWEP49 10/06/92	3
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During a hydrostatic examination, the system engineer identified a service water leak in the 10 inch discharge expansion joint on the 'B' emergency diesel generator jacket water heat exchanger (1 of 2). This failure made the 'B' diesel inoperable, but did not affect the plant which was shutdown for a refueling outage. The root cause of the failure is unknown, but localized corrosion contributed to the failure. Contract mechanics removed the inconel expansion joint and replaced it with a carbon steel spool piece. The spool piece will be monitored by NDE examination to monitor the corrosion rate. The system was leak tested satisfactorily then returned to service. This piping is due to be replaced during refuel 3, 1994. 92W004501 rcb

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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VESSEL	CALVERT CLIFFS 2	CBG	RCS	2PZVPZR21 08/05/88	1
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While the unit was down for refueling an inservice inspection revealed evidence of coolant leakage from a number of pressurizer vessel heater penetrations. Due to the severity of the cracks the system was declared inoperative. If the unit had been at power it would have been shut down, the outage was extended to make repairs. The cause of the leakage was found to be stress corrosion cracks due to over expanding the sleeves before welding them when the pressurizer was being built. The sleeves were removed, new type sleeves installed by welding, and the vessel put in service.

VESSEL	CONNECTICUT YANKEE 1	CBH	RCS	E-1 04/13/82	1
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During a 10 year inservice inspection of reactor internals, personnel found broken bolts from the core barrel to thermal shield connection on the bottom of the reactor pressure vessel. System and plant operation were not affected as the system was shut down for a refueling outage. Flow induced vibration caused movement of the core barrel leading to failure of the bolts at the core barrel thermal shield connections. The thermal shield was permanently removed after design analysis showed it was no longer required.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 11/01/82	1
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Inspection of control rod drive housing showed evidence of leaking. Intergranular stress corrosion of the CRD stub tube at the site of stub tube to CRD housing weld caused cracks that allowed reactor coolant to escape between reactor vessel and CRD housing. CRD housing was expanded by rolling into the base metal of the vessel wall closing off the path of leakage. Vessel was hydrostatically tested before startup.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/05/85	1
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During plant refueling outage, an inservice inspection of control rod drive housings 30-07 and 30-47 at their stub tubes in reactor vessel NR02 revealed evidence of leakage of reactor coolant at their stub tube heat affected zones, specifically at their stub tube to control rod drive housing welds. The cause of the failure was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. This was caused by heat and age or cyclic fatigue of the weld material. Each control rod drive housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall in order to close the path of leakage. The vessel was ultrasonically tested and hydrostatically tested and returned to service.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 06/13/85	1
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During plant shutdown for refueling, an inservice inspection of control rod drive housing 34-07 at the reactor vessel, NR02, stub tube revealed evidence of leakage of reactor water at the stub tube heat affected zone, specifically at the stub tube to control rod drive housing weld. The cause of the leakage was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. This was caused by heat and age or cyclic fatigue of the weld material. The control rod drive housing was rolled into the reactor vessel stub tube penetration in order to expand the housing into the base material of the vessel wall, thus closing the path of leakage. The heat affected zone was ultrasonically tested and hydrostatically tested and returned to service.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 01/17/87	1
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With the unit shut down, a visual examination of the under vessel control rod drive penetration stub tubes revealed leakage. Two streams of liquid were leaking in excess of the 10 drops per second acceptance criteria. The leak was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. Intergranular stress corrosion cracking is a result of heat, age and cyclic fatigue of the weld material. The defective stub tube was rolled. Rolling expands the housing into the base material of the vessel wall, closing the path of leakage. After rolling, the vessel wall was ultrasonically and hydrostatically tested.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 02/25/88	1
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With the unit in an extended outage, an inservice inspection of the control rod drive housings 46-27 and 50-19 at their stub tubes in reactor vessel nr02 revealed evidence of leakage of reactor water at their stub tube heat affected zones specifically at their stub tube to control rod drive housing welds. This would have degraded system operation. Plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. This was caused by heat and age or cyclic fatigue of the weld material. Each control rod drive housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall in order to close the path of leakage. The vessel was then checked for leaks and returned to service (WR 134342, 134343).

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 06/04/89	1
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With the unit in an extended outage, an inservice inspection of the control rod drive housing 02-19, 30-51, and 34-15 at their stub tubes in reactor vessel NR02 revealed evidence of leakage of reactor water at their stub tube heat affected zones, specifically at their stub tube to control rod drive housing welds. This would have degraded system operation. Plant operation was unaffected. The cause of the failure was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. This was caused by heat and age or cyclic fatigue of the weld material. Each control rod drive housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall in order to close the path of leakage. The vessel was then checked for leaks and returned to service (WR 183855, 183856, 183857).

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 08/30/91	1
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With the plant in cold shutdown, an inservice inspection of the control rod drive housing 10-15 at its stub tube in reactor vessel NR02 revealed evidence of leakage of reactor water at its stub tube heat affected zones, specifically at its stub tube to control rod drive housing welds, this would have degraded system operation. Plant operation was unaffected. The cause of the failure was attributed to intergranular stress corrosion cracking at the control rod drive housing to reactor stub tube weld. This was caused by heat and age or cyclic fatigue of the weld material. The control rod drive housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall in order to close the path of leakage. The vessel was then checked for leaks and returned to service. (WR199441)

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 04/15/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 18-19 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 1 drop per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of the failure was attributed to intergranular stress corrosion cracking caused by heat, aging, and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas cause cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 201379, was written to repair the leaking stub tube, the drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system returned to service.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 04/15/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 42-27 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 1 drop per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking caused by heat, aging and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas caused cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 201378 was written to repair the leaking stub tube. The drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system returned to service.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/05/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 34-31 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 10 drops per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking caused by heat, aging and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas caused cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 199798 was written to repair the leaking stub tube. The drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system returned to service.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/05/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 38-11 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 1 drop per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking caused by heat, aging and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas cause cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 203234 was written to repair the leaking stub tube, the drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path, the drive was reinstalled. A system hydro verified no leaks and the system was returned to service.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/25/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 06-31 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 1 drop per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking caused by heat, aging and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas cause cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 202155 was written to repair the leaking stub tube. The drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system was returned to service.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Svs</u>	<u>Component ID/ Date</u>	<u>Stty Class</u>
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VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/30/92	1
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With the plant in a forced outage, a special drywell inspection of control rod drive (CRD) housing 26-47 at its stub tube location in reactor vessel NR02 revealed evidence of reactor water leaking less than 8 drops per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, plant operation was unaffected. The cause of failure was attributed to intergranular stress corrosion cracking caused by heat, aging and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas caused cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 203093 was written to repair the leaking stub tube. The drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system was returned to service.

VESSEL	NINE MILE POINT 1	CBA	34/CRD	NR02 05/30/92	1
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With the unit in a forced outage, a special drywell inspection of control rod drive (CRD) housing 22-51 at its stub tube location in the reactor vessel nr02 revealed evidence of reactor water leaking less than 16 drops per minute at the heat affected zones, specifically at the stub tube to CRD housing welds. This would degrade system operation. However, the plant operation was unaffected. The cause of the failure was attributed to intergranular stress corrosion cracking caused by heat, aging, and cyclic fatigue of the CRD housing to reactor stub tube weld. Corrosion of these stress areas cause cracks resulting in a leak through the stub tube and into the annulus between the stub tube, vessel assembly and the CRD housing. WR 203092 was written to repair the leaking stub tube, the drive was removed, the CRD housing was rolled into its reactor vessel stub tube penetration in order to expand the housing with the base metal of the vessel wall thus closing the leakage path. The drive was reinstalled, a system hydro verified no leaks and the system was returned to service.

VESSEL	PILGRIM 1	CBA	0200	201-1 05/15/84	1
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On 5/15/84 with the plant shutdown for refueling outage 06 and the reactor vessel defueled, inspection of the reactor vessel riser nozzles revealed cracks. The cracks were discovered by penetrant testing which was being performed in preparation of reinstallation of the nozzle safe ends (part of recirculation pipe replacement). The cause has been attributed to intergranular stress corrosion cracking. (Reference LER 293-83063 update, 10CFR50.73a.2.ii, significant material welding defect). The nozzle was remachined. The nozzle crack indications were ground out. The nozzle was then reweld buttered and successfully penetrant tested. The vessel was successfully hydrotested on 12/1/84. (FM84-67). (MR84-2-45). (MR84-2-10).

VESSEL	POINT BEACH 2	CBH	RCS	2-T1 11/12/90	1
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During refueling primary system hydro tests, operators discovered a leak on the pressurizer manway cover ; via the manway gasket. This event extended refueling work approximately one day and delayed unit startup, and degraded the system. The manway gasket failed due to an uneven application of torque when the manway was reinstalled after the refueling. The pressurizer manway gasket was replaced. The primary system hydro test was completed and the system restored to operation.

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
VESSEL	PRAIRIE ISLAND 1	CBH	1RC	UNIT1 03/15/76	1
Seal weld leaks on unit one reactor vessel head penetrations at inst. port columns 34 and 37 and at CRDM plugs 2, 3 and 4. Unknown-weld related. Reactor head weld leaks were ground out and rewelded with reactor at refueling shutdown. Welds checked out ok at operating pressure.					
VESSEL	PRAIRIE ISLAND 2	CBH	2RC	UNIT2REACT. 01/03/76	1
Seal weld leaks on unit 2 reactor vessel head at inst. port columns 34 and 37 and at CRDM plugs U3 and U5. Unknown-weld related. Reactor head weld leaks were ground out and rewelded with reactor at cold shutdown and reactor level below vessel flange welds then checked at 2335 psig.					
VESSEL	SAN ONOFRE 1	CBH	RC	C-1 11/30/85	1
Unit in refueling. Reactor vessel thermal shield was inspected for degraded supports in response to finding core barrel thermal shield support hardware loose on top of fuel at Connecticut Yankee. Found 5 of 6 flexures broken, 3 of 30 support block bolts broken and 1 of the dowel pins in a lower support block had a broken tack weld. Thermal shield supports were degraded but plant operation was not affected. The root cause was primarily wear at the 4 limiter key locations, causing increased loads on the lower support blocks. This allowed relative motion between the core barrel and the thermal shield to increase enough so that fatigue failure occurred in 5 of the 6 flexures. MO89120482/87112715. Committed to NRC to repair the thermal shield in cycle 11 refueling. Removed thermal shield, old flexures, limiter keys, and support blocks from the vessel. Revised design flexures, limiter keys and support blocks were installed with new fasteners (piece parts). The thermal shield was inspected, installed in the vessel, and returned to service.					
VESSEL	TROJAN 1	CBH	RCS	T-201 09/13/84	1
During reactor coolant system inspection, found leak on incore instrument fitting at the seal table. While attempting to tighten the fitting, the fitting failed, causing loss of coolant accident (LOCA). Previous installations had reduced the fitting's ferrule grip due to a slightly tapered tube end. Attempting to adjust under system pressure caused failure and leak. Cooled down and depressurized and drained down RCS. Applied temporary weld repair to stop leak. New design tubing coupling will be installed next outage. Trained maintenance personnel on tubing fitting assembly/disassembly work.					

<u>Comp</u>	<u>Plant</u>	<u>System Code</u>	<u>Util Sys</u>	<u>Component ID/ Date</u>	<u>Sfty Class</u>
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VESSEL	ZION 1	CBH	RC	RC001 12/15/89	1
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During a scheduled containment walkdown of the primary systems by the in-service inspection group, a small leak was discovered on the reactor head instrumentation column seal (CONOSEAL). The leakage was a small bubbling at the upper and lower seal on one column and at the lower joint on a second. The unit was at hot standby and had to be shutdown to repair the leak. The cause of the event was determined to be faulty gaskets. The lower gaskets had scratches on the outside diameter sealing surface, and the upper gaskets had irregularities on the outside diameter sealing surface. The possibilities are: the faults were missed during installation inspection, and/or the faults were caused during installation. The unit was immediately brought down to cold shutdown, Mode 5. All of the port CONOSEAL gaskets were then replaced with new gaskets like in kind to those recommended by manufacturer. The unit was then brought back up to temperature and pressure. The port columns were inspected for leakage, and found to be satisfactory. (DVR 22-01-89-180)

VESSEL	DRESDEN 2	CBA	203	201-1 01/10/86	2
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In refueling during a remote visual inspection of the reactor vessel internals, it was discovered that the top 4-1/2 in. of the intermediate radiation monitor IRM-18 dry tube plunger was missing (part of the traversing incore probe system). The operation of IRM-18 was normal during the previous cycle. Irradiation assisted stress corrosion cracking. See GE SIL #409 for details. Component is GE part #886D380G001. The lost part of the plunger will be found or a lost parts analysis will be performed. A work request was written to replace the broken dry tube.

VESSEL	DRESDEN 3	CBA	203	3-201-1 11/16/84	2
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During a refueling outage, a crack indication was found in jet pump beam assembly #11 via remote ultrasonic testing examination. Stress corrosion cracking. Replaced the beam assembly with a new one.

Appendix C
Excerpts from ISI Summary Reports

Appendix C

Excerpts from ISI Summary Reports

Plant A

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1.0 Summary of Inservice Inspection

This report describes the Inservice Inspection of Duke Power Company's Oconee Nuclear Station Unit 3 during the 1992 Refueling Outage (also referred to as Outage 13, which is in the Third Inspection Period of the Second Inspection Interval).

Included in this report are the final Inservice Inspection Plan, the inspection results for each item, a summary for each category of examination and corrective action taken when unacceptable conditions were found. In addition, there is a section included for repairs and replacements required since March 23, 1991.

1.1 Class 1 Inspection

The Class 1 Inservice Inspection included examinations on the Reactor Vessel Circumferential Head Welds, Reactor Vessel Head to Flange Weld, Reactor Vessel CRDM J-Groove Nozzle Welds and Incore Monitoring Partial Penetration Nozzle Welds. In addition, examinations were performed on the Reactor Vessel Closure Head Nuts, Studs, Threads in Flange, Closure Washers and Bushings.

Examinations were performed on Pressurizer Lower Head to Heater Bundle Circumferential Seam Weld, Shell to Heater Bundle Long Seam Welds and Manway Studs, Nuts and integrally-welded attachments.

Examinations were performed on Steam Generators 3A and 3B Upper Head to Tubesheet Welds and Nozzle to Vessel Welds, Steam Generator 3A Lower Manway Studs and Nuts. Steam Generator 3B Upper Manway Studs and Nuts, Upper and Lower Head Inspection Cover Bolting also received examination.

Dissimilar Metal Butt Welds in the Reactor Coolant System were examined. Bolting in Core Flood Valve 3CF-14, Low Pressure Injection Valve 3LP2 and Pressurizer Relief Valve 3RC-4 also received examinations.

Piping welds in Reactor Coolant, Low Pressure Injection and High Pressure Injection Systems were examined.

Visual examinations were performed on the Class 1 Pressure Boundary during system leakage tests. Also, visual examinations were performed on Class 1 Component Supports of the High Pressure Injection and Low Pressure Injection Systems and on the Reactor Pressure Vessel Support Skirt.

The Inconel 600 tubing in Steam Generators 3A and 3B was inspected by eddy current during Outage 13. The results are shown in Section 5 of this report. A detailed description of the Oconee 3 Refueling Outage 13 eddy current inspection is provided in the "Eddy Current Examination Report Oconee Unit 3, August 1992 Refueling Outage 13" on file in the Corporate Office in Charlotte,

Reportable indications were found on the Class 1 Inspections shown in this section. Inspections and evaluation data for each reportable indication found on Class 1 Inspections is included in Section 5 of this report.

A detailed description of each inspection is found in the final Inservice Inspection Plan in Section 3 of this report. Results of each examination are found in Section 4.

1.2 Class 2 Inspection

The Class 2 Inspections included examinations on Steam Generator 3A Shell to Shell Weld, Shell to Nozzle Belt Welds, Low Pressure Injection Coolers 3A and 3B Inlet and Outlet Nozzle to Shell Welds, Steam Generators 3A and 3B Outlet Nozzle to Shell Welds and Core Flood Tanks 3A and 3B Outlet Nozzle to Head Welds. Also, examinations were performed on vessel integrally-welded attachments of Steam Generator 3B Feedwater Header Support and integrally-welded attachments of the Main Steam, Low Pressure Injection and Reactor Building Spray Systems and on Steam Generator 3B Feedwater Header.

Examinations were performed on circumferential butt welds of the Low Pressure Injection, Reactor Building Spray, Spent Fuel Cooling, High Pressure Injection, Auxiliary Feedwater, Main Feedwater, Main Steam and Component Cooling Systems. Also, longitudinal welds of the Low Pressure Injection, Reactor Building Spray, Main Steam and Spent Fuel Cooling Systems. In addition, branch connection welds of the Main Feedwater and Main Steam Systems were examined.

Visual examinations were performed on the Class 2 Pressure Boundary during System Leakage and Hydrostatic Tests. Also, visual examinations were performed on Class 2 Component Supports of the Main Steam, Main Feedwater, Low Pressure Injection, Reactor Building Spray and Component Cooling Systems.

Reportable indications were found on the Class 2 Inspections shown in this section. Inspection and evaluation data for each reportable indication is included in Section 6 of this report.

A detailed description of each inspection is found in the final Inservice Inspection Plan in Section 3 of this report. Results of each examination are found in Section 4.

1.3 Augmented Inspection

Augmented Inspections at Oconee 3 consisted of Reactor Coolant Pump Flywheel examination, Thermal Stress Piping Examinations, Auxiliary Feedwater Header Water Hammer Examinations, and Pressurizer Sensing/Sampling Safe-End Nozzle Examinations. A

detailed description of each examination is found in the final Inservice Inspection Plan in Section 3 of this report. Results are found in Section 4.

1.4 Alternate Inspection

Alternate inspections at Oconee 3 were performed on welds 3-50-45-3 and 3-50-45-4 (Item Numbers E03.001.001 and E03.001.002 respectively), and on Reactor Coolant Pumps 3A2 and 3B1 Flange.

A detailed description of each examination is found in the final Inservice Inspection Plan in Section 3 of this report. Results are found in Section 4.

1.5 Identification Numbers

Owner: Duke Power Company, 422 S. Church Street, Charlotte, NC 28201
Plant: Oconee Nuclear Station, Highway 130/183, Seneca, SC 29679
Plant Unit: 3
Owner Certification of Authorization: N/A
Commercial Service Date: 12/16/74

<u>Item</u>	<u>Manufacturer Or Installer</u>	<u>Manufacturer or Installer Serial No.</u>	<u>State or Province No.</u>	<u>National Board No.</u>
Reactor Vessel	Babcock & Wilcox	620-0004-51-52	N/A	N-105
Steam Generator "A"	Babcock & Wilcox	620-0004-55	N/A	N-107
Steam Generator "B"	Babcock & Wilcox	620-0004-55	N/A	N-108
Pressurizer	Babcock & Wilcox	620-0004-59	N/A	N-106

1.6 Authorized Nuclear Inservice Inspector(s)

Name: M. B. Chapman *MB Chapman 12-3-92*

Employee: The Hartford Steam Boiler
Inspection and Insurance Company
200 Ashford Center North
Suite 300
Atlanta, GA 30338

5.0 Class 1 Inspection Results

Examinations were performed during Outage 13 on the Reactor Vessel, Pressurizer, Steam Generators 3A and 3B, Piping Welds of the Reactor Coolant, Low Pressure Injection and High Pressure Injection Systems, Valve Bolting and Supports.

5.1 Reactor Vessel

The Reactor Vessel Closure Head Ring to Closure Head Cap Weld received an ultrasonic examination, the Vessel Head to Flange Weld received an ultrasonic and a magnetic particle examination. No reportable indications were found.

Sixty-nine (69) Reactor Vessel Closure Head J-Groove CRDM Nozzle Penetration Welds and fifty-two (52) Reactor Vessel Incore Monitoring Nozzle Partial Penetration Welds received a visual examination. No reportable indications were found.

Thirty (30) Reactor Vessel Closure Head Nuts received a magnetic particle examination. Thirty (30) Reactor Vessel Closure Studs received an ultrasonic and a magnetic particle examination. No reportable indications were found.

The threads in Reactor Vessel Flange at 180° to 0° received an ultrasonic examination. No reportable conditions were found.

Reactor Vessel Closure Washers and Bushings for Studs Holes 1-9, 25-35, and 51-60 received a visual examination. No reportable conditions were found.

The Reactor Vessel Support Skirt Weld received a visual examination. No reportable conditions were found.

5.2 Pressurizer

The Pressurizer Lower Head to Heater Bundle Circumferential Seam Weld and Heater Bundle Longitudinal Seam Weld located at "X" and "Y" Axis received ultrasonic examinations. No reportable indications were found.

Twelve (12) Pressurizer Manway Studs received an ultrasonic examination. No reportable indications were found.

Four (4) Pressurizer Support Lug Welds located at "X-axis between "X-Y" axis, at "Y" axis and between "Y-Z" axis received a magnetic particle examination. No reportable indications were found.

Twelve (12) Pressurizer Manway Nuts, Bushings and Washers received a visual examination. No reportable conditions were found.

5.3 Steam Generators

Steam Generators 3A and 3B Upper Head to Tubesheet Welds received an ultrasonic examination. Reportable indications in Steam Generator 3B Upper Head to Tubesheet Weld, 3GSB-WG58-1 (Item Number B02.040.003), detected during Outage 6 were re-examined by ultrasonics during Outage 13 per ASME Section XI, Paragraph IWB-2420(b). A Babcock and Wilcox Flaw Evaluation, Number 32-1218901-00 was performed and it was determined that the previously reported flaw indications remains acceptable under the criteria of ASME Section XI, Paragraph IWB-3612. A copy of the evaluation report is included in Section 10 of this report. A copy of the inspection data sheets is included in this section of the report. No other reportable indications were found.

Steam Generator 3A Inlet and Outlet Nozzle and Steam Generator 3B Inlet Nozzle to Vessel Welds received ultrasonic examinations. No reportable indications were found.

Steam Generator 3A Lower Head Manway Studs and Nuts, Steam Generator 3B Upper Head Manway Studs and Nuts and Upper and Lower Head Inspection Cover Bolting received visual examinations. No reportable conditions were found.

5.4 Valves

Bolting for Valves 3CF-14, 3LP-2 and 3RC-4 received a visual examination. No reportable conditions were found.

5.5 Piping

Dissimilar Metal Butt Welds:

Seven (7) dissimilar metal butt welds, nominal pipe size four inches and over received an ultrasonic and a dye penetrant examination. Reportable indications were found on weld 3PIB2-7 (Item Number B05.050.007A). The indications were removed, weld repaired and re-examined. The re-examination revealed no indications. Refer to Problem Investigation Report 3-092-0332 included in Section 9 of this report. The inspection data sheets for the reportable item are included in this section of the report. The inspection sample was extended to include four (4) additional welds per ASME Section XI Paragraph IWB-2430(a). No other reportable indications were found.

Two (2) dissimilar metal butt welds, nominal pipe size less than four inches, received a dye penetrant examination. No reportable indications were found.

Similar Metal Butt Welds:

Nominal pipe size four inches and greater:

Eight (8) circumferential butt welds received an ultrasonic and a dye penetrant examination. No reportable indications were found.

Four (4) longitudinal welds received nd ultrasonic and a magnetic particle examination. No reportable indications were found.

Nominal pipe size less than four inches:

Twenty (20) circumferential welds received a dye penetrant examination. No reportable indications were found.

Two (2) branch pipe connection welds received a magnetic particle examination. No reportable indications were found.

Pipe Supports:

Sixteen (16) Class 1 Component Supports received a visual examination as required by ASME Section XI Article IWF-2000. No reportable conditions were found.

5.6 Steam Generator Tubing

The Inconel 600 tubing in the two Babcock and Wilcox once-through steam generators (OTSG's) "A" and "B" at Duke Power Company's Oconee Unit 3, were examined by eddy current techniques during August 1992. The tubing in the generators measures 0.625" nominal outside diameter X 0.037" nominal wall thickness. The examinations were performed by Babcock and Wilcox Nuclear Services (BWNS) Company during the 13th refueling outage.

Approximately 60% of the tubes in steam generators "A" and "B" were inspected full length with magnetic-bias bobbin coil probes. Also, all of the previously installed sleeves were examined with a combination bobbin/crosswound probe. Numerous tubes and rolled plugs were examined using Zetec's motorized rotating pancake coil (MRPC) probe.

The following is a summary of the results of the bobbin coil/MRPC eddy current examination for each generator.

OTSG A

- Eddy Current examinations began on 8/8/92 and were completed on 8/27/92.
- 74 tubes were removed from service by plugging.

- 9,325 tubes were examined by bobbin coil for the full length of the tube.
- 25 tubes contained indications of 40% thru-wall (TW) or greater reported by bobbin coil.
- 90 tubes contained indications of 20 to 39% TW reported by bobbin coil.
- 147 previously installed sleeves were examined with a combination bobbin/crosswound probe.
- 142 tubes in the lane/wedge region were examined at the 15th tube support plate (TSP) and upper tubesheet secondary face (UTSF) intersections using Zetec's MRPC.
- 856 special interest tubes were examined using Zetec's MRPC.
- 125 rolled plugs were examined from the hotleg and 42 rolled plugs from the coldleg using Zetec's MRPC.
- 2 tubes that were re-expanded approximately 5" from the tube ends in the coldleg tubesheet and one in hotleg tubesheet were examined using Zetec's MRPC.
- 147 upper rolls in sleeves were examined using Zetec's MRPC.

OTSG B:

- Eddy Current examinations began on 8/8/92 and were completed on 8/26/92.
- 52 tubes were removed from service by plugging.
- 9,326 tubes were examined by bobbin coil for the full length of the tube.
- 17 tubes contained indications of 40% TW or greater reported by bobbin coil.
- 79 tubes contained indications of 20 to 39% TW reported by bobbin coil.
- 97 previously installed sleeves were examined with a combination bobbin/crosswound probe.
- 222 tubes in the lane/wedge region were examined at the 15th TSP and UTSC intersections using Zetec's MRPC.
- 700 special interest tubes were examined using Zetec's MRPC.

- 92 hotleg rolled plugs and 33 coldleg rolled plugs were examined using Zetec's MRPC.
- 2 tubes that were re-expanded approximately 5" from the tube end were examined using Zetec's MRPC. Of these 2 tubes, 1 was in the hotleg tubesheet and the remaining one was in the coldleg tubesheet.
- 96 upper rolls in sleeves were examined using Zetec's MRPC.

A detailed description of the Oconee 3 Refueling Outage 13 eddy current inspection is provided in the "Eddy Current Examination Report Oconee Unit 3 August 1992 Refueling Outage 13" on file at the Corporate Office in Charlotte, North Carolina.

5.7 System Leakage Test

The Class 1 Pressure Boundary was subjected to a system leakage test as required by ASME Section XI, Article IWB-5000. No reportable conditions were found.

5.8 Class 1 Repairs and Replacements

Repairs and replacements for work performed from March 24, 1991 to September 25, 1992 are itemized in Section 11 of this report.

7.0 Augmented Inspection and Alternate Examination Results

Augmented inspections were performed on Reactor Coolant Pumps 3A1, 3A2, 3B1 and 3B2 Flywheels, Thermal Stress Piping (NRC Bulletin 88-08), Auxiliary Feedwater Header (PSC 21-82) Water Hammer concern, and Pressurizer Sensing/Sampling Nozzle Safe-Ends.

Alternate examination were performed on Pressurizer Relief Valve Piping Welds and on Reactor Coolant Pumps 3A2 and 3B1.

7.1 Reactor Coolant Pump Flywheels

Reactor Coolant Pumps 3A1, 3A2, 3B1 and 3B2 Flywheel received an ultrasonic examination. No reportable indications were found.

7.2 Thermal Stress Piping

Eleven (11) welds received an ultrasonic examination as required by NRC Bulletin 88-08. No reportable indications were found.

7.3 Auxiliary Feedwater Header (PSC-21-82)

Thirteen (13) welds received a magnetic particle examination in accordance with the Auxiliary Feedwater Header (PSC-21-82) Water Hammer concern. No reportable indications were found.

7.4 Pressurizer Sensing/Sampling Nozzle Safe-Ends

Seven (7) nozzle to safe-end welds received a dye penetrant examination. No reportable indications were found.

7.5 Alternate Examination

A dye penetrant examination was performed on piping welds between Pressurizer Relief Valves 3RC4 and 3RV-67.

Welds 3 and 4 (ISI Item Number E03.001.001 and E03.001.002 respectively) on Duke Power Company Isometric Drawing System 50 ISO 45, located between Pressurizer Relief Valves 3RC4 and 3RV-67, received a dye penetrant examination, in lieu of the applicable hydrostatic test specified in ASME Section XI, Table IWB-2500-1 Category B-P. These examinations were performed in accordance with the Inservice Inspection Program, Request for Relief ONS-005, submitted to the Nuclear Regulatory Commission. A copy of the request is included in Section 10 of this report. A copy of the inspection data sheets is included in this section of the report.

No reportable indications were found.

Today's Date : 08/25/92
Time : 10:28
Page No.: 1
PIP Serial No.: 3-092-0371

PIR Serial No.: 3-092-0371
LER Serial No.:
Other Rpt. No.:

OCONEE NUCLEAR STATION
Problem Investigation Process
Problem Investigation Form

I. Problem ID

Occurred Time/Date : / / Disc Time/Date : 08/19/92

Unit(s): 3	Status at Time Occurred:	Unit 1	Unit 2	Unit 3
	Mode:	N/A	N/A	6
	% Power:	N/A	N/A	N/A

Unit Status Remarks: Refueling Outage #13

System(s) Affected: 1.) MS 2.) 3.) 4.)

System Definitions:
MS = Main Steam

Problem Associated With Work Order No.:

Location of Problem - Bldg: R Elev: 825' Column Line: A CAV.
Location Remarks: STEAM GEN. 3A SHELL WELDS

Description and Cause of Problem:

During Ultrasonic Examination of ISI Item # C01.010.002, ID #3SGA-WG8-2, and ISI Item # C01.010.001, ID # 3SGA-WG8-1 intermittent indications were revealed 360 degrees on ID running parallel with the weld. The indications are unacceptable in accordance with ASME Section XI- 1980 Edition Winter 1980 Addenda Table IWB-3511-1.

Originated by: TJCOLEM Section: CES Date: 08/19/92

Revised by: TJCOLEM Section: CES Date: 08/19/92

Immediate Corrective Actions:

NONE

Originated by: TJCOLEM Section: CES Date: 08/19/92

Corrective Action Work Order No.:

Problem Identified By: TJCOLEMA	Date: 08/19/92	Section: CES
Problem Entered By : TJCOLEM	Date: 08/19/92	Section: CES

II. Significance

Event Code(s) : 1.) 24 2.) 3.) 4.)
Event Code Definitions:

Today's Date : 08/25/92
Time : 10:28
Page No.: 2
PIP Serial No.: 3-092-0371

PIR Serial No.: 3-092-0371
LER Serial No.:
Other Rpt. No.:

**OCONEE NUCLEAR STATION
Problem Investigation Process
Problem Investigation Form**

24 = Preventive Maintenance Related

Init. Cause Code(s): 1.) X 2.) 3.) 4.)
Initial Cause Code Definitions:
X = Other / Unknow Causes

Resp. Section(s) : 1.) CES 2.) 3.) 4.)
Section Code Descriptions:
CES = Component Engineering

Is the Problem Significant? Y If yes, Significance Code: 5
Significance Code = Assistance from Engineering
If Yes, enter the PIR, IIR, & LER No. if known and as applicable.
PIR No.: 3-092-0371 LER No.: Special Rpt No.:

Screening Remarks:

Originated by: TKROYAL Section: CES Date: 08/25/92

Screened By: TKROYAL Date: 08/25/92 Section: CES

III. Problem Evaluation

Resp. Section(s) : 1.) CES 2.) 3.) 4.)
Section Code Descriptions:
CES = Component Engineering

System(s) Affected: 1.) MS 2.) 3.) 4.)
System Definitions:
MS = Main Steam

Component Type : =

Manufacturer : =

Equipment ID No. :

Root Cause Code(s): 1.) 2.) 3.) 4.)

Today's Date : 08/25/92
Time : 10:28
Page No.: 3
PIP Serial No.: 3-092-0371

PIR Serial No.: 3-092-0371
LER Serial No.:
Other Rpt. No.:

OCONEE NUCLEAR STATION
Problem Investigation Process
Problem Investigation Form

Root Cause(s) of Problem:

Proposed Resolution From CES :

Originated by: TKROYAL Section: CES Date: 08/25/92

Is a Document or Physical Change to the Station Required?

If Yes, enter the Change Do . No.s:

Change Doc. No. 1:

Change Doc. No. 2:

Change Doc. No. 3:

Remarks:

IV. Approval

Approved By: Date: / / Section:

End of the Document for PIP No.: 3-092-0371
This Status of this PIP No. is : PIR

PROBLEM RESOLUTION FORM

Problem Investigation Serial No. 3-092-0371

PROBLEM EVALUATION

Root Cause of Problem: These indications have been monitored by Duke Power Co. and B+W and do not appear to be growing. The root cause can not be determined but the indications do not appear to be service induced.
Basil W. Carney

Cause Codes

☐ ☐ ☐ ☐ ☐ ☐

Proposed Resolution: B+W will evaluate the indications using the requirements of ASME Section II, IWB-3612.

This ~~work~~^{BWC} work was completed and the indications are acceptable without repairs (see PIR 3-092-0334 for a copy of B+W's report.)
^{BWC} 9/24/92 B+W's report is attached.

Physical change to Station or Controlled Document requiring exempt change or NSM: ☐ Yes ☒ No

* If yes, send corrective action to Systems Engineering.

Group(s) Involved For Resolution: Component Engineering

Evaluated By: Basil W. Carney Date: 9/24/92
 Supervisory Concurrence: Ted K. Royal Union City Date: 9/24/92

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Plant B

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Repair/Replacements	Page 8

Tables

Table I	Examinations Completed by Category
Table II	Limited Examinations
Table III	Snubber Testing Summary
Table IV	Significant Indications

Appendices

- A. NIS-1 Owners Data Report
- B. NDE Results
- C. Repair/Replacement Listing
NIS-2 Data Reports

SUMMARY

Approximately 77% of the first Inservice Inspection Interval requirements have been completed. Nuclear Regulatory Commission (Commission) augmented examinations in high energy lines, core spray spargers, jet pump beams, feedwater spargers, snubber testing and in lines covered by Generic Letter 88-01 were completed per commitments.

One surface indication was found that required corrective measures. This indication was blended out and re-examined with acceptable results. Additional surface examinations of similar welds did not reveal any additional unacceptable indications.

A suspected IGSCC flaw, discovered at R6, was re-examined. Results did not show any significant change from R6. The analysis performed at R6 for continued operation is still valid. The weld will be examined again in R8.

EXAMINATION RESULTS

This report summarizes the results of Inservice Inspection (ISI) of ASME Class 1,2 and 3 components and supports performed at Washington Public Power Supply System (Supply System) Nuclear Plant No. 2 (WNP-2) between September 30, 1991 and July 18, 1992. Both General Electric (GE) and Supply System personnel performed the examinations. During this period, WNP-2 completed the seventh scheduled refueling outage, RF92A (R7).

This report includes the NIS-1 Owner's Report of Inservice Inspection. A copy can be found in Appendix A.

The following table lists the ISI inspection periods and outages for the first interval.

<u>Inspection Period</u>	<u>Refueling¹ Outage</u>	<u>From</u>	<u>To</u>
1		<u>12/13/84</u>	<u>09/15/88</u>
	RF86A (R1)	03/31/86	06/10/86
	RF87A (R2)	04/13/87	06/25/87
	RF88A (R3)	05/02/88	06/27/88
2		<u>09/16/88</u>	<u>09/30/91²</u>
	RF89A (R4)	04/28/89	06/30/89
	RF90A (R5)	04/21/90	08/07/90
	RF91A (R6)	04/15/91	09/30/91
3		<u>10/01/91²</u>	<u>12/13/94</u>
	RF92A (R7)	04/18/92	07/18/92
	RF93A (R8)	04/15/93	06/15/93
	RF94A (R9)	04/15/94	06/15/94

(1) Assumes one refueling or maintenance outage each year. Actual timing of the spring outage for R8 and R9 may vary due to BPA's hydroelectric capacity or outage scope.

(2) R6 outage was extended

ISI SUMMARY REPORT RF92A

The ISI examinations are specified in ASME Section XI and required by 10CFR50.55a. In addition, the following examinations were performed to meet augmented Nuclear Regulatory Commission requirements.

- o pipe break exclusion areas (high energy lines penetrating containment, but not within ASME Section XI examination boundary)
- o IGSCC (intergranular stress corrosion cracking) detection in stainless steel welds, based on Generic Letter 88-01.
- o visual examination of Core Spray Spargers and supply piping in RPV

The ASME Section XI examinations comply with the 1980 Edition, Winter 1980 Addenda upgraded as follows:

- o IWA-2300(a)(1), 1983 Edition, Winter 1983 Addenda
- o Code category C-F, 1983 Edition, Winter 1983 Addenda
- o IWF-3400, 1980 Edition Winter, 1981 Addenda
- o IWB-3600, 1986 Edition no Addenda (to comply with GL88-01)

Documentation supporting this Summary Report is included in the WNP-2 ISI Program Plan or is located in the WNP-2 Operations File. Table I lists all ASME Section XI and augmented examinations completed during R7. Appendix B contains a more detailed summary of the results by system and ISI drawing. The ISI drawings referenced are included in the ISI Program Plan previously submitted to the Commission.

The examinations, tests, repairs and replacements were witnessed or verified by Authorized Nuclear Inspector-Inservice (ANI-I) D.E. Hoggarth. He is employed by Factory Mutual Systems, a subsidiary of Arkwright Mutual Insurance Company, Norwood, Massachusetts.

COMPONENTS EXAMINED

The following components were examined:

<u>Component</u>	<u>Manufacturer</u>	<u>Serial Number</u>	<u>National Board No.</u>
RPV	CBIN Nuclear 2700 Channel Ave Memphis, TN	T-45	8
RCIC-V-65	Velan Engineering Co. 2125 Ward Montreal QUE	0334	NA
RHR-V-41C	Velan Engineering Co. 2125 Ward Montreal QUE	67	NA
RHR-V-50B	Velan Engineering Co. 2125 Ward Montreal QUE	414	NA

<u>Component</u>	<u>Manufacturer</u>	<u>Serial Number</u>	<u>National Board No.</u>
RRC-V-67A	Atwood & Morrill Co. Inc. 285 Canal St Salem, Mass	3-336	NA
RRC-V-67B	Atwood & Morrill Co. Inc. 285 Canal St Salem, Mass	4-336	NA

PIPING EXAMINATIONS

- o Twenty-five (25) safe end welds (GL 88-01 category "D" and "G") were examined using the GE SMART 2000 inspection system. No unacceptable indications were found. See Generic Letter 88-01 summary below under "AUGMENTED EXAMINATIONS" for further details.
- o Two hundred four (204) class 1 and 2 examinations were completed by UT, PT or MT methods. One weld, 12RRC(1)-N2K-4, had a rejectable surface indication. The indication was blended out and reexamination resulted in an acceptable result. Additional welds of this type were examined with acceptable results. The indication was attributed to silicon inclusions or porosity in the original weld that was opened up during operational thermal cycling.

RPV EXAMINATIONS

- o Interior visual examinations were performed by General Electric. Various augmented and Section XI category B-N-1 VT-3 items were examined. See augmented examination section below for further details. No unacceptable indications were found.
- o Nozzle to vessel welds were examined by GE using their GERIS examination system and manual scanning. Top head welds were completed manually by GE and Supply System personnel. The RPV weld ultrasonic examinations were performed in accordance with ASME Section XI 1980 Edition Winter 1980 Addendum and NRC Regulatory Guide 1.150 Revision 1 alternate method. No unacceptable indications were detected.

Full ASME Section XI Code coverage of the nozzle to vessel welds could not be obtained. The nozzle blend radius configuration prevents full coverage. Table II lists the nozzle type and the percent examined.

AUGMENTED EXAMINATIONS

The Supply System performed augmented examinations per the ISI Program Plan section 5.3, "Mandatory Augmented Inservice Inspection".

- o High energy lines that penetrate containment (section 5.3.1)
Dye penetrant or ultrasonic examination of 14 welds in the high energy line augmented examination program was performed. No reportable indications were found. This brings the total welds examined in this category to 57.

o Core spray sparger and supply piping (section 5.3.5)

A visual examination of the core spray spargers and their supply piping was performed per the requirements of IE Bulletin 80-13, "Cracking in Core Spray Sparger". The examination was performed using an underwater closed circuit TV system capable of resolving a 0.001 inch diameter wire in-situ. No relevant indications were observed.

o Jet pump beam visual (section 5.3.3)

The jet pump beams were visually examined using the VT-1 technique. No reportable indications were found.

o Generic Letter 88-01 (section 5.3.4)

General Electric examined 25 IGSCC category "D" and "G" welds (safe-ends) and 1 category "F" weld using the GE SMART 2000 automated system. The category "F" weld, 20RRC(6)-8, had a reportable indication detected at R6. The results of this exam determined that the flaw size had not changed significantly. The analysis performed at R6 for continued operation is still valid. The results of this examination and analysis for continued operation was submitted to the Commission for review and approval for continued operation. (ref. letter GO2-92-123, dated May 14, 1992) The Commissions approved operation for one more cycle. (ref. letter dated June 25, 1992, TAC M83721) This weld will be examined again at R8 to determine any change in flaw size. No indications were recorded for the category "D" and "G" welds. In addition 18 category "B" and 4 category "A" welds were examined using either the SMART 2000 system or manually. No indications were recorded.

Status of GL 88-01 commitment:

Category (Total #) ¹	Required within 6 yrs ¹	Required within 10 yrs ¹	WNP-2 Status thru R7 (After 4 yrs) ¹
A (54)	12%	25%	47% ²
B (147)	25%	50%	21%
D (8)	100% ³	100% ³	100% ³
G (17)	100% ⁴	100% ⁴	100% ⁴
F (1) ⁵	100% ⁶	100% ⁶	100% ⁶

Notes

- 1 WNP-2 commitment began at R4.
- 2 WNP-2 requirements exceed GL 88-01 because of ASME Section XI requirements.
- 3 100% of welds require examination every 2 refueling outages
- 4 All welds require examination every refueling outage. All welds classified as "G" in R7 are now reclassified as "D".
- 5 This category "B" weld was reclassified to category "F" at R6.
- 6 100% of welds require examination every refueling outage.

WNP-2 has met or is on schedule with its GL 88-01 commitments for the first six years.

o Feedwater spargers (section 5.3.2)

The feedwater spargers were visually examined by VT-1 technique per Supply System ISI Program Plan and FSAR commitments. No reportable indications were found.

o Snubber testing (section 6.5)

An initial sample of 37 snubbers was selected from the WNP-2 general population of 564 safety related snubbers. These snubbers were randomly selected by computer sub-routine which is part of the Snubber Test and Examination Program (STEP). The selected snubbers were then reviewed to determine if the sample was representative as required by Technical Specification 4.7.4.e.

Testing of snubbers was performed using portable testing devices "Validators". These devices were supplied by the snubber manufacturer. Testing results summary are found in Table II.

Snubber MD-1285-14D S/N 4047 passed the functional test, but drag was over 2%. To preclude further service life degradation it was replaced with another tested snubber S/N 2528 deleted from RCIC-970S under MWR AR6826.

The next testing is required within 18 months.

LIMITED EXAMINATIONS

Full ASME Section XI required coverage of the examination volume or surface could not be accomplished on 23 welds. Table II is a summary of the coverage obtained. The remaining volume or surface will be examined at a future outage or if necessary relief will be requested.

SIGNIFICANT INDICATIONS

Significant indications found during ISI examinations are summarized in Table IV. All significant indications were evaluated. Rejectable indications/items were repaired or replaced. Evaluation and/or re-examination data sheets are attached to the original data reports.

REPAIRS AND REPLACEMENTS

Eight (8) significant repair and replacement activities were performed during the RF92A refueling outage: 1) Modified valves RRC-V-67A and RRC-V-67B, 2) Installed new multiholed orifices in the HPCS system, 3) Installed new multiholed orifices in the FPC system, 4) Rerouted RHR return lines, 5) Replaced LPRM's, 6) Overhauled and replaced CRD's, 7) Replaced disc inserts and nozzles for five (5) main steam relief valves and replaced four (4) main steam relief valves, 8) Continuation of the snubber optimization program and 9) Modified vent/drain/test connections. A listing and summary of these and all other repairs and replacements accomplished between October 1, 1991 and July 18, 1992 are provided in Appendix C.

1) Reactor Recirculation Cooling (RRC) System

Installed new modified valve bonnets and internals for valves RRC-V-67A and RRC-V-67B to improve resistance to flow induced wear failures. Performed VT-3 visual examinations for ISI on the internal accessible surfaces of the new modified bonnets and the existing valve bodies.

2) High Pressure Core Spray (HPCS) System

Installed new multiholed orifices HPCS-RO-8 and HPCS-RO-9 to prevent vibration caused by cavitation in the HPCS full flow test return line to the suppression pool. Performed surface and volumetric examinations on the new welds for ISI (Preservice Inspections - PSI).

3) Fuel Pool Cooling (FPC) System

Installed new multiholed orifices FPC-RO-5A and FPC-RO-5B to reduce cavitation during operation of the RHR - Assist mode of FPC system.

4) Residual Heat Removal (RHR) System

Rerouted RHR return lines from hydrogen recombiner skids to eliminate the potential for water to back up the drain lines to CAC recombiner skids when the RHR system test return line is operated.

5) LPRM's

Replaced four (4) Local Power Range Monitoring (LPRM) incore assemblies.

6) CRD's

Overhauled eleven (11) Control Rod Drives (CRD's) and replaced thirty one (31) Control Rod Drives (CRD's).

7) Main Steam (MS) System

Replaced disc inserts and nozzles for four (4) spare main steam relief valves with Serial Numbers N63790-00-0046, N63790-00-0050, N63790-00-0052 and N63790-00-0056. These four (4) spare main steam relief valves were installed in place of the existing main steam relief valves in the plant as follows :

<u>EPN No</u>	<u>Serial No</u>
MS-RV-1C	N63790-00-0046
MS-RV-1D	N63790-00-0050
MS-RV-3B	N63790-00-0052
MS-RV-4B	N63790-00-0056

Replaced disc insert and nozzle for one (1) additional spare main steam relief valve with Serial Number N63790-00-0053. This spare relief valve was installed in place of MS-RV-3B, S/N N63790-00-0052 because this relief valve was inoperable during plant start up.

8) Snubber Optimization Program

As part of the Supply System's effort to reduce the number of safety related snubbers at WNP-2, twenty three (23) snubbers were replaced with rigid struts and forty six (46) snubbers were deleted. The new struts received Preservice Inspections (PSI) after installation.

9) Vent/Drain/Test Connections

Modified eight (8) vent/drain/test connections for RHR, LPCS and HPCS systems to reduce susceptibility to fatigue induced failures at socket welds.

TABLE IV
Significant Indications

Data sheet No.	Identification No.	Description	Remarks
1RPV-160	CRD HOUSING 42-59 BLT	CRD Housing Bolting	Corrosion pitting attack along shank. Metalurgical evaluation acceptable for use.
1RIU-047	6RCIC(1)-72	ELL TO PIPE	125% DAC due to ID geometry.
1HPU-015	16HPCS(1)-42	ELL TO PIPE	125% DAC due to ID geometry.
1LPU-028	12LPCS(1)-2	PIPE TO ELL	130% DAC due to ID geometry.
1LPM-016	24LPCS(2)-8	PIPE TO ELL	Linear indication found by MT. Acceptable by full thickness UT of indication surface area as permitted by ASME Section XI IWC-3514.
1HV-0223	LPCS-902N	SPRING	Spring can alignment off approximately 5 degree. Evaluated as acceptable.
1MSV-143	4MS(12)-1BD	FLANGE BOLTING	Bolts damaged during disassembly.
1RRP-098	12RRC(1)-N2K-4	PIPE TO SAFE-END	Rounded PT indication. Blended out and re-examined by PT acceptable.
1RRU-166	20RRC(6)-8	PIPE TO VALVE	Re-sized indication found at R6 and found insignificant difference in size from previous outage. Will re-size at R8.
1HV-0220	SW-430	STRUT	Race displaced from paddle 1/4" on pipe side. Evaluated as acceptable.

TABLE IV
Significant Indications

Data sheet No.	Identification No.	Description	Remarks
1HV-0219	SW-941N	BOX SUPPORT	Baseplate nuts corroded. Evaluated as acceptable. Nuts replaced and coated.
1HV-0218	SW-940N	BOX SUPPORT	Baseplate nuts corroded. Evaluated as acceptable. Nuts replaced and coated.
1HV-0217	SW-933N	BOX SUPPORT	Baseplate nuts corroded. Evaluated as acceptable. Nuts replaced and coated.
1HV-0216	MS-324	SPRING	Clamp bolting loose. Evaluated as acceptable.

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APPENDIX C

ASME SECTION XI REPAIR AND REPLACEMENT LISTING NIS-2 OWNER'S REPORTS

This appendix summarizes all ASME Section XI repairs and replacements work completed between October 1, 1991 and July 18, 1992. The status of the NIS-2 Owner's Report is stated for each repair and replacement work performed. For repairs and replacements completed during R-7 but for which an NIS-2 Owner's Report has not been issued yet, the NIS-2 Owner's Report will be issued in the next ISI Summary Report.

PLAN NO	MWR NO	COMPONENT NUMBER / WORK DESCRIPTION	DESC OF COMP	R&R REPORTED IN
2-0629	AR 0381	Replaced body to bonnet nuts for valves RHR-V-27A and RHR-V-27B	Valve	RF92A Summary Report
2-0653	AR 2188	Replaced disc insert and nozzle for relief valve MS-RV-1B	Relief Valve	RF92A Summary Report
2-0654	AR 2189	Replaced relief valve MS-RV-1C with spare S/N N63790-00-0120	Relief Valve	RF92A Summary Report
2-0655	AR 2190	Replaced disc insert and nozzle for relief valve MS-RV-2B	Relief Valve	RF92A Summary Report
2-0656	AR 2191	Replaced disc insert and nozzle for relief valve MS-RV-2C	Relief Valve	RF92A Summary Report
2-0657	AR 2192	Replaced relief valve MS-RV-2D with spare S/N N63790-00-0124	Relief Valve	RF92A Summary Report
2-0658	AR 2193	Replaced disc insert for relief valve MS-RV-4D	Relief Valve	RF92A Summary Report
2-0659	AR 2194	Replaced disc insert and nozzle for relief valve MS-RV-5B	Relief Valve	RF92A Summary Report
2-0660	AR 2195	Replaced disc insert and nozzle for relief valve MS-RV-5C	Relief Valve	RF92A Summary Report
2-0666	AR 5711	Reinstalled and welded access cut out plates on shell for CCH-CR-1A	Heat exchanger	RF92A Summary Report
2-0667	AR 3178	Replaced evaporator tubes for CCH-CR-1A	Heat exchanger	RF92A Summary Report
2-0675	AR 3722	Modified test connection for valves RFW-V-45A and RFW-V-45B	Piping	RF92A Summary Report
2-0676	AR 3723	Modified test connection for valves RFW-V-119 and RFW-V-120	Piping	RF92A Summary Report
2-0677	AR 3721	Modified test connection for valves RFW-V-44A and RFW-V-44B	Piping	RF92A Summary Report
2-0678	AR 3724	Modified test connection for RHR-V-161A and RHR-V-162A	Piping	See Plan No 2-0741
2-0679	AR 3720	Modified drain connection for valves HPCS-V-21 and HPCS-V-22	Piping	RF92A Summary Report
2-0680	AR 1095	Modified test connection for valves RFW-V-121 and RFW-V-122	Piping	RF92A Summary Report
2-0691	AR 0904	Body to bonnet seal weld and replaced stem/disc for valve MS-V-1	Valve	RF92A Summary Report
2-0692	AR 0905	Body to bonnet seal weld and replaced stem/disc for valve MS-V-2	Valve	See Plan No 2-0807
2-0693	AR 0906	Body to bonnet seal weld and replaced stem/disc for valve MS-V-5	Valve	RF92A Summary Report
2-0701	AR 3567	Replaced bolting material for flange joints on Dwg RCIC-659-27.28	Piping	RF92A Summary Report
2-0712	AR 3795	Modified new bonnet assemblies for valves RRC-V-67A and RRC-V-67B	Valves	RF92A Summary Report
2-0713	AR 3793	Installed new parts valve RRC-V-67A	Valve	RF92A Summary Report
2-0714	AR 3794	Installed new parts valve RRC-V-67B	Valve	RF92A Summary Report
2-0717	AR 5341	Replaced condenser tubes for CCH-CR-1A	Heat exchanger	RF92A Summary Report
2-0722	AR 5544	Cut and rewelded weld Dwg SW-508-1.8	Piping	RF92A Summary Report
2-0723	AR 5544	Cut and rewelded weld Dwg CCH-101-1.6	Piping	RF92A Summary Report
2-0724	AR 7052	Rerouted instrument piping PI(1)-4S-X78B	Tubing	RF92A Summary Report
2-0725	AR 6985	Installed piping for set pressure verification for MSRV's	Piping	RF92A Summary Report
2-0726	AR 7156	Modified test connection for valves HPCS-V-57 and HPCS-V-718	Piping	RF92A Summary Report
2-0727	AR 6116	Installed valve RCIC-V-215 for pressure gage RCIC-PI-11	Piping	RF92A Summary Report
2-0730	AR 7157	Fabricated restriction orifice for HPCS-RO-8 and HPCS-RO-9	Piping	RF92A Summary Report
2-0731	AR 7157	Fabricated spool pieces for HPCS-RO-8 and HPCS-RO-9	Piping	RF92A Summary Report
2-0732	AR 7156	Installed HPCS-RO-8 and HPCS-RO-9	Piping	RF92A Summary Report
2-0737	AR 7283	Modified test connection for HPCS-V-37 and HPCS-V-38	Piping	See Note 1
2-0738	AR 7298	Modified test connection for RHR-V-157A and RHR-V-158A	Piping	RF92A Summary Report
2-0739	AR 7285	Modified test connection for LPCS-V-37 and LPCS-V-38	Piping	See Note 1
2-0740	AR 7299	Modified test connection for RHR-V-157C and RHR-V-158C	Piping	See Note 1
2-0741	AR 7300	Modified test connection for RHR-V-161A and RHR-V-162A	Piping	RF92A Summary Report
2-0750	AR 5998	Replaced mechanical seal for pump RRC-P-1B	Pump	RF92A Summary Report
2-0751	AR 6615	Installed valve for RHR-PI-18	Tubing	RF92A Summary Report
2-0752	AR 6032	Installed cathodic protection of spray pond siphon piping	Piping	RF92A Summary Report
2-0753	AR 6386	Modified test connection for RHR-V-161A and RHR-V-162A	Piping	RF92A Summary Report
2-0754	AR 6388	Modified test connection for HPCS-V-37 and HPCS-V-38	Piping	RF92A Summary Report
2-0755	AR 6387	Modified test connection for RHR-V-157B and RHR-V-158B	Piping	RF92A Summary Report
2-0756	AR 6389	Weld built up for valves RHR-V-163 and RHR-V-164	Piping	RF92A Summary Report
2-0757	AR 3556	Replaced disc insert and nozzle for relief valve S/N 63790-00-0056	Relief Valve	RF92A Summary Report
2-0758	AR 3557	Replaced disc insert and nozzle for relief valve S/N 63790-00-0050	Relief Valve	RF92A Summary Report
2-0759	AR 4043	Replaced parts for relief valve S/N 63790-00-0052	Relief Valve	RF92A Summary Report
2-0760	AR 4044	Replaced disc insert and nozzle for relief valve S/N 63790-00-0046	Relief Valve	RF92A Summary Report
2-0761	AR 5675	Installed flanges for DCW-HX-1A2 drain line	Piping	RF92A Summary Report
2-0763	AR 5675	Repaired corroded areas on cover plate for DCW-HX-1A2	Heat Exchanger	RF92A Summary Report
2-0764	AR 6489	Deactivated valve SW-V-223A	Valve	RF92A Summary Report
2-0765	AR 6488	Deactivated valve SW-V-223B	Valve	RF92A Summary Report
2-0766	AR 5797	Modified piping for relief valve SW-RV-1A	Piping	RF92A Summary Report
2-0767	AR 5798	Modified piping for relief valve SW-RV-1B	Piping	RF92A Summary Report
2-0769	AR 6317	Body to bonnet seal weld for valve SW-V-931A	Valve	RF92A Summary Report
2-0770	AR 1359	Assembled stem disc and disc piston assemblies for MSIV	Valve	RF92A Summary Report
2-0773	AR 5668	Reinstalled drain connection with valve SW-V-834B and CCH-V-702B	Piping	RF92A Summary Report
2-0774	AR 7159	Installed pipe cap for connection with valve HPCS-V-718	Piping	RF92A Summary Report
2-0776	AR 6912	Installed pipe cap for connection with valve HPCS-V-70	Piping	RF92A Summary Report
2-0777	AR 7431	Replaced body to bonnet bolting material for valve RHR-V-50B	Valve	See Note 1
2-0778	AR 6913	BDC-88-0138-2D work for JCI support B-220-1077-145	Support	See Note 1

PLAN NO	MWR NO	COMPONENT NUMBER / WORK DESCRIPTION	DESC OF COMP	R&R REPORTED IN
2-0779	AR 6913	BDC-88-0138-2D work for PDM reinforcement structural steel	Supports	See Note 1
2-0780	AR 7435	Modified test connection for RHR-V-157B and RHR-V-158B	Piping	See Note 1
2-0781	AR 7436	Modified test connection for RHR-V-163 and RHR-V-164	Piping	RF92A Summary Report
2-0782	AR 7425	Modified test connection for HPCS-V-713 and HPCS-V-714	Piping	See Note 1
2-0783	AR 7564	Fabricated restriction orifice for FPC-RO-5A and FPC-RO-5B	Piping	RF92A Summary Report
2-0784	AR 7531	Fabricated spool pieces for FPC-RO-5A and FPC-RO-5B	Piping	RF92A Summary Report
2-0785	AR 7617	Installed FPC-RO-5A and FPC-RO-5B	Piping	RF92A Summary Report
2-0788	AR 7788	Modified connection for valve FPC-V-811	Piping	RF92A Summary Report
2-0792	AR 6826	Machined under sized pins	Supports	RF92A Summary Report
2-0793	AR 6826	Replaced snubbers with rigid struts for MS supports	Supports	RF92A Summary Report
2-0794	AR 6826	Replaced snubbers with rigid struts for MSRV supports	Supports	RF92A Summary Report
2-0795	AR 6826	Replaced snubbers with rigid struts for RCIC supports	Supports	RF92A Summary Report
2-0796	AR 6826	Replaced snubber with rigid strut for HPCS support	Support	RF92A Summary Report
2-0797	AR 7791	Replaced Local Power Range Monitoring (LPRM) in core assemblies	RPV	RF92A Summary Report
2-0798	AR 7748	Rerouted RHR return line from recombiner skid, RHR - Loop A	Piping	RF92A Summary Report
2-0799	AR 7749	Rerouted RHR return line from recombiner skid, RHR - Loop B	Piping	RF92A Summary Report
2-0800	AR 7781	Modified flanges for RHR - Loop A	Piping	RF92A Summary Report
2-0801	AR 7782	Modified flanges for RHR - Loop B	Piping	RF92A Summary Report
2-0802	AR 7617	Modified connection for valve FPC-V-612	Piping	RF92A Summary Report
2-0804	AR 7834	Modified hydrogen recombiner CAC-HR-1A skid piping	Piping	RF92A Summary Report
2-0805	AR 7835	Modified hydrogen recombiner CAC-HR-1B skid piping	Piping	RF92A Summary Report
2-0806	AR 4935	Replaced stem/disc assembly for valve MS-V-1	Valve	See Note 1
2-0807	AR 4944	Body to bonnet seal weld for valve MS-V-2	Valve	See Plan No 2-0849
2-0808	AR 7066	Installed J pipe cap for connection with valve RCIC-V-603	Piping	See Note 1
2-0809	AR 7825	Installed spring steps for relief valve CIA-RV-5A	Relief valve	RF92A Summary Report
2-0810	AR 7825	Installed spring steps for relief valve CIA-RV-5B	Relief valve	RF92A Summary Report
2-0813	AR 6532	Replaced relief valve MS-RV-1C with spare S/N N63790-00-0048	Relief valve	RF92A Summary Report
2-0814	AR 6533	Replaced relief valve MS-RV-1D with spare S/N N63790-00-0050	Relief valve	RF92A Summary Report
2-0815	AR 6534	Replaced relief valve MS-RV-3B with spare S/N N63790-00-0052	Relief valve	RF92A Summary Report
2-0816	AR 6535	Replaced relief valve MS-RV-4B with spare S/N N63790-00-0058	Relief valve	RF92A Summary Report
2-0817	AR 7595	Replaced disc insert for relief valve CCH-RV-2A	Relief valve	See Note 1
2-0818	AR 7044	Removed socket weld pipe cap and installed threaded pipe cap	Piping	RF92A Summary Report
2-0819	AR 8426	Installed test port for relief valve HPCS-RV-35	Relief Valve	See Note 1
2-0820	AR 6609	Removed scratches on the disc seating surface for valve MS-V-41	Valve	See Note 1
2-0821	AR 7478	Replaced bolting material for flanged joints on Dwg RCIC-659-27.28	Piping	RF92A Summary Report
2-0822	AR 8363	Replaced parts for valve SLC-V-4B	Valve	RF92A Summary Report
2-0823	AR 5436	Replaced bolting material for OG-HX-1B	Heat Exchanger	See Note 1
2-0824	AR 8531	Rerouted instrument line PI(1)-4S-x115 on Dwg D-220-1.2-X115	Tubing	RF92A Summary Report
2-0825	AR 6300	Replaced U bolts for support MS-954N	Support	RF92A Summary Report
2-0826	AR 5436	Plugged tube for OG-HX-1B	Heat Exchanger	See Note 1
2-0828	AR 7478	Replaced bolting material for RPV Nozzle N8	Vessel	RF92A Summary Report
2-0829	AR 8630	Removed vent valves SW-V-168A and SW-V-169A	Piping	RF92A Summary Report
2-0831	AR 8660	Rerouted piping/tubing for LPCS-FE-2	Piping/Tubing	RF92A Summary Report
2-0832	AR 6353	Replaced rear disc assembly for valve CVB-V-1GH	Valve	See Note 1
2-0833	AR 1672	Installed pipe nipple and pipe cap for valve RWCU-V-605	Piping	RF92A Summary Report
2-0834	AR 4989	Replaced disc, nozzle and stem for relief valve SLC-RV-29A	Relief valve	RF92A Summary Report
2-0835	AR 4990	Replaced disc, nozzle and stem for relief valve SLC-RV-29B	Relief valve	RF92A Summary Report
2-0836	AR 7141	Replaced bolting material for decon flanged joint	Piping	RF92A Summary Report
2-0837	AR 8646	Body to bonnet seal weld for valve SLC-V-7	Valve	RF92A Summary Report
2-0841	AR 7735	Machined gasket seating surface on the bonnet for valve RFW-V-10A	Valve	RF92A Summary Report
2-0842	AR 8755	Removed PT indication from RPV nozzle safe end N2-K-SE-330, Weld 5	Piping	RF92A Summary Report
2-0843	AR 3793	Reinstalled bonnet vent connection for valve RRC-V-67A	Piping	RF92A Summary Report
2-0844	AR 3794	Reinstalled bonnet vent connection for valve RRC-V-67B	Piping	RF92A Summary Report
2-0845	AR 8856	Replaced safe end for valve CIA-SPV-1A	Valve/Piping	RF92A Summary Report
2-0846	AR 8760	Body to bonnet seal weld for valve CIA-V-56B	Valve	RF92A Summary Report
2-0849	AR 4944	Replaced valve MS-V-2	Piping	RF92A Summary Report
2-0850	AR 8737	Replaced bolting material for cover plate bolted joint for DCW-HX-1B2	Heat Exchanger	RF92A Summary Report
2-0851	AR 8960	Machined surface defects on valve plug for valve RHR-FCV-64B	Valve	See Note 1
2-0852	AR 8990	Replaced wedge for valve RHR-V-27B	Valve	See Note 1
2-0853	AR 9103	Replaced relief valve HPCS-RV-35 inlet piping material	Piping	See Note 1
2-0854	AR 8986	Replaced relief valve RHR-RV-1A	Relief valve	RF92A Summary Report
2-0855	AR 8988	Replaced disc for relief valve RHR-RV-25A	Relief valve	See Note 1
2-0856	AR 9283	Body to bonnet seal weld for valve RHR-V-143A	Valve	RF92A Summary Report

PLAN NO	MWR NO	COMPONENT NUMBER / WORK DESCRIPTION	DESC OF COMP	R&R REPORTED IN
2-0857	AR 9525	Replaced disc insert and nozzle for relief valve S/N 63790-00-0053	Relief Valve	RF92A Summary Report
2-0858	AR 9529	Replaced relief valve MS-RV-3B with spare S/N N63790-00-0053	Relief valve	RF92A Summary Report
2-0861	AR 7534	Installed pipe cap for connection RCIC-V-103	Piping	RF92A Summary Report
2-0862	AR 7566	Installed pipe cap for connection LPCS-V-36	Piping	RF92A Summary Report
2-0863	AR 7567	Installed pipe cap for connection LPCS-V-59	Piping	RF92A Summary Report
2-0864	AR 7286	Installed pipe cap for connection LPCS-V-55	Piping	RF92A Summary Report
2-0866	AR 9978	Removed and reinstalled elbow shown on Dwg SW-1525-1	Piping	RF92A Summary Report
N/A	AR 6626	Deleted snubbers	Supports	RF92A Summary Report
N/A	AR 5782	Replaced Control Rod Drives (CRD's)	CRD	RF92A Summary Report
N/A	AR 0572	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 0641	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5867	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5869	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5870	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5873	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5878	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5879	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5889	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 5891	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
N/A	AR 8522	Overhauled and replaced part(s) for Control Rod Drive (CRD)	CRD	RF92A Summary Report
Note 1	Work completed but NIS-2 Owner's Report not yet issued			

1R10
OUTAGE SUMMARY REPORT

The regularly-scheduled ISI examinations performed during 1R10 are listed in the attached table. 1R10 was the second and last refueling outage in the second period of the second 10-year Interval. The attached table includes the non-destructive examinations other than the pressure tests and associated VT-2 exams. The pressure tests are summarized below.

Pressure Tests

All remaining second-period pressure tests, as required by the ASME Code and by the ISI Program, were performed during this outage. In addition, the following 10-year ISI hydrostatic tests were completed during Cycle 10:

- I. Both Service Water loops including supply and suction headers except the following components within the supply and return isolation valves:
 1. Emergency diesel generator coolers (E20B1 & E20B2)
 2. Auxiliary Building Electrical Rooms Emergency Chillers (E-175 & E-177)
 3. Auxiliary Building Decay Heat Removal Unit Coolers (VUC-1A, 1B, 1C, & 1D).
- II. Emergency Feedwater Pump suction piping.

Component Supports (Hangers)

The current ISI Program requires inspection of 100% of the component supports (hangers) during each period. This requirement is carried over from the first 10-year ISI Program. Consequently, a very large number of hangers were inspected during this outage. All but a few of the hangers were found to be in satisfactory condition and these few hangers had only minor problems. Most of the problems had to do with loose bolts and nuts, probably due to vibration associated with an operating plant. A few hanger welds were found to be undersized and one spring hanger was discovered to be carrying no load. All of these conditions were either corrected or resolved by engineering evaluation.

In the case of the spring hanger with no load, additional supports were inspected and found to be satisfactory. Therefore, the condition appears to have been an isolated case. Due to oversight, four of the hangers (24-082, 24-092, 50-020, and 53-010) with loose bolts and nuts did not receive a visual inspection after the maintenance was performed to tighten the loose parts. Consequently, a Condition Report (CR-1-92-486) was issued to have the reinspections performed during the next refueling outage. The

CR also requires that the process of assigning post-maintenance testing requirements be reviewed for adequacy.

Intergranular Stress Corrosion Cracking (IGSCC)

In addition to the ASME Code-required examinations, an augmented program of examinations was performed on thin-wall stainless steel piping connected to the Borated Water Storage Tank (BWST). This piping was selected for examination due to the discovery of a cracked weld during plant restart from refueling outage 1R9. A few months after completion of 1R9, another crack was discovered in thin-wall stainless steel piping in the Spent Fuel Pool Cooling (SFP) System. Since both of these cracks were in similar piping carrying the same borated water, a common cracking mechanism was believed to exist. These cracks were the basis for initiation of Condition Reports CR-1-91-131 and CR-1-91-199.

The basis for the selection of welds to include in the augmented program was NRC Bulletin 79-17 which addresses Intergranular Stress Corrosion Cracking (IGSCC) in stainless steel piping in Pressurized Water Reactors (PWRs). The original scope for this outage was 26 welds, which was a 10% sample of thin-wall piping welds that were not already in the ISI Program. The BWST piping was examined using ultrasonic techniques specially designed for the detection of IGSCC in thin-wall stainless steel. During the examinations, one weld was found to have a small IGSCC indication. The same weld that had the crack in 1R9 still contained a small crack that had not been detected by radiography performed in 1R9 and subsequently had not been repaired. The small indication is believed to have existed from the original startup timeframe of the early 1970s. The entire weld was removed, along with the adjacent elbow, and two new welds were added. Two additional welds, one on either side of the cracked weld, were added to the scope of the augmented program. In addition, one of the original 26 welds was found to be partially obstructed due to a pipe hanger; consequently, another adjacent weld was added to the scope. Thus, the final scope of this augmented program was 29 welds. It was determined from the examination results, which were reviewed by NRC Region IV inspectors, that IGSCC is not a significant concern at ANO-1 at this time. On-going IGSCC examinations will be performed as part of our augmented ISI Program.

In order to determine the root cause of the cracking, a cracked weld from the SFP system was sent to Southwest Research Institute (SwRI) for metallurgical evaluation. The SwRI report (06-4722-001, contract A-2071) concluded that the crack was indeed IGSCC and that the probable cause was sulfides. The sulfides are believed to have been introduced into the borated water systems during the original startup of ANO-1 in the 1970s. The original design of the plant included a Sodium Thiosulfate Tank in the Reactor Building Spray (BS) System. During startup, this tank either leaked or was dumped into the BS System which communicates directly with the BWST which, in turn, feeds the SFP System. The sulfates in the Sodium Thiosulfate solution break down

into sulfides which then cause cracking in sensitized stainless steel welds. During the early 1970s, ANO-1 and other B&W plants with Sodium Thiosulfate Tanks had cracking problems in thin-wall stainless steel piping systems. These results were reported to the NRC who then initially issued IE Circular 76-06 and later issued IE Bulletin 79-17.

The SFP System also received a visual (VT-2) inspection to determine if any other IGSCC cracks existed. One indication was found on the body of valve SF-8B. It is a through-wall seepage, but is not located in the weld or heat-affected zone. Therefore, the seepage is probably not caused by IGSCC, but may simply be a casting defect in the valve body. Another Condition Report (CR-1-92-0077) has been generated to track this indication. This minor seepage can be repaired in a non-outage timeframe and poses no reactor safety or spent fuel pool cooling problems.

Pure Water Stress Corrosion Cracking (PWSCC)

Other augmented examinations occurred during 1R10. One of these involved an ultrasonic examination of a repair weld installed during 1R9. When ANO-1 was heating up at the end of 1R9, a leak was detected in one of the three upper level taps on the Pressurizer. The level tap is a 1-inch Inconel pipe inserted through a drilled hole in the side of the Pressurizer. The Inconel pipe is secured in place with a single weld on the inside diameter of the Pressurizer. After discovery of the leak, the unit was shut down and cooled down in order to repair the weld. Rather than go inside the Pressurizer, a repair weld was installed on the outside. In making this repair, the Inconel pipe had to be cut off within the Pressurizer wall and a new section of pipe welded in place with the exterior repair weld. A gap of approximately 0.060-inch had to be left between the old and new sections of piping. This gap could allow borated water to come in contact with the carbon steel Pressurizer wall. Even though calculations qualified the repair until 1R11, ANO committed to perform an ultrasonic examination of the suspect area during 1R10 to determine if any corrosion of the base material was taking place. This examination was performed by SwRI personnel and witnessed by ANO NDE personnel. No corrosion was detected during the operating cycle between 1R9 and 1R10. The Pressurizer upper level tap repair weld is tracked in the ISI Program as A-10173. The SwRI report is filed in the ANO Calc Room as report number 91-R-1017-02.

The root cause of the cracked weld is presently believed to be Pure Water Stress Corrosion Cracking (PWSCC), which is a cracking mechanism that can attack Inconel materials in a very pure water environment such as the pressurizer steam space in the Reactor Coolant System (RCS). In order to determine if any other Inconel materials in the RCS are experiencing cracking problems, another augmented program was performed during 1R10. Ten areas on the Pressurizer that contain Inconel, including the repair weld, were visually (VT-2) inspected after the plant had been heated up to normal operating pressure and temperature at the end of the outage. No leakage was detected. This same exam is scheduled to be repeated each outage for the near future.

Other Piping Weld Indications

Additional stainless steel piping cracks were found near the end of the refueling outage. Workers in the Reactor Building noticed that one of the welds in the Pressurizer Safety Valve Discharge lines ("tailpipe") appeared to be dripping water. The horizontal runs of piping on top of the Pressurizer were ultrasonically examined. Five of the 19 welds were found to have crack-like indications. Consequently, the horizontal runs of piping on both safety valve discharges were replaced with new piping. One of the cracked welds has been sent to SwRI for metallurgical evaluation to determine the root cause of the cracking. The evaluation is currently underway. It should be noted that although this safety valve discharge piping was originally built to ASME Class 2 standards, it does not have a nuclear safety function. However, the piping will be inspected in the future as part of the augmented ISI Program. This activity is also being tracked by CR-1-92-0244.

Reactor Coolant Pumps "A" and "B" Re-examinations

During previous outages, flaws were detected in the casing welds of Reactor Coolant Pumps (RCPs) A and B. Per requirements in Section XI of the ASME Code, these flaws must be reexamined three times in the ten years following their discovery. These flaws, which are believed to be slag inclusions left over from the original fabrication of the pumps, were reexamined during this refueling outage. The examinations were performed by B&W using an improved version of the same ultrasonic equipment utilized several outages ago to find and size these indications. The inclusions did not grow in size from the previous inspections performed during 1R8. In fact, due to the refinements made in the UT equipment, the slag inclusions appeared to shrink in size. From this examination, it is concluded that the RCP casing welds remain in a satisfactory condition. These examinations are contained in the ISI Program and in this report as 43-001A, 44-001, 44-001A, and 44-002.

Reactor Coolant Pump "D" Flywheel

During the coast-down from power to the refueling outage, the bearings in RCP D were damaged. Consequently, during the outage the pump and motor were disassembled. Since ANO-1 has a Tech Spec requirement to perform a ten-year examination on the RCP Flywheels, the flywheel on RCP D was examined while the motor was disassembled and sitting on the motor stand. The motor stand provides better access to the flywheel and therefore a more thorough exam can be performed. The flywheel received an ultrasonic and a magnetic particle (MT) exam. The examinations revealed no recordable indications.

Reactor Coolant Pump "D" Casing Welds

ANO-1 committed to the NRC in a letter dated October 27, 1988 (1CAN108805) to radiograph C and D RCP casing welds any time that either pump is disassembled. However, since RCP D was not planned to be disassembled, but rather discovered during the outage to be damaged, it was decided that the unexpected radiography would be a hardship on the plant. At the time of the outage, ASME had recently published Code Case N-481 which allows a nuclear plant owner to substitute an engineering evaluation for the radiographic inspection. ANO approached the NRC to obtain approval of the use of the Code Case. The NRC granted approval for the use of the Code Case for 1R10. The engineering evaluation was performed by a contractor, Structural Integrity Associates, with assistance by ANO personnel. Code Case N-481 was submitted to the NRC after the outage in letter dated June 12, 1992 (1CAN069201). If the NRC approves of the conclusions drawn in the Code Case, then all future radiographs of the RCP casing welds on all four RCPs will also be deleted as stated in ANO letter 1CAN069201.

Reactor Pressure Vessel Studs and Nuts

Magnaflux, a manufacturer of magnetic particle testing (MT) materials, issued a recall notice in 1991 to their customers. Some of the aerosol cans of fluorescent MT materials may not have contained the proper amounts of suspended solids. ANO-1 had used some of the questionable materials during the previous outage, 1R9, on 20 Reactor Pressure Vessel (RPV) studs and 10 RPV nuts. These studs and nuts were reexamined during 1R10 using new MT materials from Magnaflux. In addition, a magnetic field indicator, sometimes referred to as a "pie gauge", was utilized to verify both field strength and direction. The examinations revealed no recordable indications. This re-examination was tracked by CR-C-91-0102.

09/03/92

Page 1

ENERGY OPERATIONS
ARKANSAS NUCLEAR ONE - UNIT 1
1R10 COMPLETED ISI EXAMINATIONS

No results

<u>CATEGORY</u>	<u>ITEM NO.</u>	<u>ISI EXAM NO.</u>	<u>COMPONENT DESCRIPTION</u>	<u>EXAMINATION AREA DESCRIPTION</u>	<u>NDE METHOD(S)</u>
B-B	B 2.11	05-008	PRESSURIZER	HTR BELT SHELL TO LO HD CIR SM	UT
B-B	B 2.11	05-009	PRESSURIZER	HTR BUNDLE SHELL HALF CIR SEAM	UT
B-B	B 2.12	05-010	PRESSURIZER	HTR BDL SHELL LONG SEAM X-AXIS	UT
B-B	B 2.12	05-011	PRESSURIZER	HTR BDL SHELL LONG SEAM Y-AXIS	UT
B-B	B 2.40	03-002	STEAM GENERATOR E24A	TUBESHEET TO LOW SHELL TRAN WD	UT
B-B	B 2.40	03-003	STEAM GENERATOR E24A	LOWER HEAD TO TUBESHEET WELD	UT
B-D	B 3.110	05-015	PRESSURIZER	PZR RELIEF NOZZ BETWEEN Z-W AXIS (NOZZLE TO VESSEL WELD)	UT
B-D	B 3.110	05-015IR	PRESSURIZER	PZR RELIEF NOZZ BETWEEN Z-W AXIS (INSIDE RADIUS SECTION)	UT
B-D	B 3.130	03-006	STEAM GENERATOR E24A	NOZZLE TO LOWER HEAD WELD	UT
B-D	B 3.140	03-007	STEAM GENERATOR E24A	NOZZLE TO HEAD INNER RADIUS	UT
B-F	B 5.50	18-001	PRESSURIZER SPRAY	NOZZLE TO PIPE SE CIRC SEAM	UT ,PT
B-F	B 5.50	24-066A	LETDOWN COOLER & DRAIN	PIPE TO SAFE END	PT
B-G-1	B 6.10	01-N-022	REACTOR VESSEL	CLOSURE HEAD NUT #22	MT
B-G-1	B 6.10	01-N-024	REACTOR VESSEL	CLOSURE HEAD NUT #24	MT

C-36

Plant D

INSERVICE INSPECTION REPORT
UNIT 1 CATAWBA 1992 REFUELING
OUTAGE 6

Location: 4800 Concord Road, York, SC 29745

NATIONAL BOARD NO. 130

COMMERCIAL SERVICE DATE: JUNE 29, 1985

Owner: Duke Power Company
422 S. Church St.
Charlotte, NC 28201

Revision 0

Prepared By:

J. J. Hogge Jr.
J. E. Cherry
G. B. Barlow

Date

12-11-92

Reviewed By:

Date

12-21-92

Approved By:

Date

12/21/92

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1.0 Summary of Inservice Inspection

This report describes the Inservice Inspection of Duke Power Company's Catawba Nuclear Station Unit 1 during the 1992 Refueling Outage (also referred to as Outage 6, which is in the Third Inspection Period of the First Inspection Interval).

Included in this report are the final Inservice Inspection Plan, the inspection results for each item, a summary for each category of examination and corrective action taken when unacceptable conditions were found. In addition, there is a section included for repairs and replacements required since June 9, 1991.

1.1 Class 1 Inspection

The Class 1 Inservice Inspection included examinations on the Reactor Vessel CRD Housing Welds, Steam Generator 1A Channel Head to Tubesheet Weld and on Pressurizer Safety Nozzle To Safe End Butt Welds. Examinations were performed on Class 1 Dissimilar Metal Butt Welds of the Upper Head Injection System, and on circumferential butt welds in the Reactor Coolant (NC), and Safety Injection (NI) Systems. Also, examinations were performed on Reactor Coolant Pump 1C Main Flange Bolting and 1NC-27 Valve Body Weld.

Visual examinations were performed on the Class 1 Pressure Boundary during System Leakage Tests. Reactor Coolant Pumps 1A, 1B, 1C and 1D Flange, Reactor Coolant Pump 1C Seal Gland and Seal Housing Bolts, Valves 1NC-1, 1NC-2 and 1NC-3 Studs and Nuts and the internal surfaces of Valves 1NC-27 and 1NI-29 also were examined. In addition, visual examinations were performed on Class 1 Component Supports of Reactor Coolant (NC), Residual Heat Removal (ND), Safety Injection (NI) Systems and on Pressurizer Support Skirt, Support Lugs, Upper Support and Lower Support Frame.

The Inconel 600 tubing in Steam Generators 1A, 1B, 1C and 1D was inspected by eddy current during Outage 6. The results are shown in the Catawba Unit 1 July 1992 Refueling Outage Eddy Current Examination Report.

Reportable indications were found on the Class 1 Inspections shown in this section. Inspection and evaluation data for each reportable indication is found in the Catawba Unit 1 July 1992 Refueling Outage Eddy Current Examination Report.

A detailed description of each inspection is found in the final Inservice Inspection Plan in Section 3 of this report. Results of each examination are found in Section 4.

1.2 Class 2 Inspection

The Class 2 Inspections included examinations on RHR Heat Exchanger 1A Inlet Nozzle "A," and Outlet Nozzle "B" Nozzle to Shell Welds, Steam Generator 1B Main Steam Nozzle to Upper Head Weld and Steam Generator 1D Auxiliary Feedwater Nozzle to Shell Weld. Examinations were performed on circumferential butt welds of the Auxiliary Feedwater (CA), Main Steam Supply to Auxiliary Equipment (SA), Main Steam (SM) and Main Steam Vent to Atmosphere (SV), Main Feedwater (CF) and Safety Injection (NI) Systems. Examinations were performed on piping integrally-welded attachments of Main Feedwater (CF) System.

Visual examinations were performed on the Class 2 Pressure Boundary during System Functional Tests. Also, visual examinations were performed on Class 2 Component Supports of Auxiliary Feedwater (CA), Main Feedwater (CF), Refueling Water (FW), Safety Injection (NI), Main Steam Supply to Auxiliary Equipment (SA) and Main Steam (SM) Systems.

A detailed description of each inspection is found in the final Inservice Inspection Plan in Section 3 of this report. Results of each examination are found in Section 4.

1.3 Augmented Inspection

Augmented Inspections at Catawba 1 consisted of Reactor Coolant Pump Flywheel examinations, Steam Generator Tubing examinations, Pipe Rupture Protection examinations, and Thermal Stress Piping examinations. A detailed description of each examination is found in the final Inservice Inspection Plan in Section 3 of this report. Results of each examination are found in Section 4.

1.4 Alternate Inspection

No alternate examinations were performed during Outage 6.

1.5 Identification Numbers

Owner: Duke Power Company, 422 S. Church Street, Charlotte, NC 28201
Plant: Catawba Nuclear Station, 4800 Concord Road, York, SC 29745
Plant Unit: 1
Owner Certification of Authorization: N/A
Commercial Service Date: 6/29/85

2.0 Status of Required Inspections

The completion status of inspections required by the 1980 ASME Section XI Code, including addenda through Winter 1981, is summarized in this section. The requirements are listed by the ASME Section XI examination category as defined in Table IWB-2500-1 for Class 1 Inspections and in IWC-2500-1 for Class 2 Inspections. Augmented inspections are also included.

Class 1 Inspections

<u>Section XI Category</u>	<u>Description</u>	<u>Inspections Required</u>	<u>Inspections Completed</u>	<u>Percentage Completed</u>	<u>Deferral Allowed</u>
B-A	Pressure Retaining Welds in Reactor Vessel	14 Welds	2.5 Welds	17.85%	Yes
B-B	Pressure Retaining Welds in Vessels Other than Reactor Vessel	8 Welds	6 Welds	75%	No
B-D	Full Penetration Welds of Nozzles in Vessels	36 Inspections	18 Inspections	50%	Partial
B-E	Pressure Retaining Partial Penetration Welds in Vessels	55 Welds	55 Welds	67%	No
B-F	Pressure Retaining Dissimilar Metal Welds	46 Welds	32.5 Welds	70.65%	No
B-G-1	Pressure Retaining Bolting Greater Than 2 Inch Diameter	370 Items	250 Items	67.56%	Yes
B-G-2	Pressure Retaining Bolting 2 Inches and Less in Diameter	*59 Connections	42 Connections	71.18%	No

*Total connections includes CRDMS that are disassembled

Class 1 Inspections (Continued)					
<u>Category</u>	<u>Description</u>	<u>Inspections Required</u>	<u>Inspections Completed</u>	<u>Percentage Completed</u>	<u>Deferral Allowed</u>
B-H	Integral Attachments for Vessels	9 Attachments	5 Attachments	55.55%	No
B-J	Pressure Retaining Welds in Piping	201 Welds	157 Welds	78.10%	No
B-K-1	Integral Attachments for Piping, Pumps and Valves	None	N/A	N/A	N/A
B-L-1	Pressure Retaining Welds in Pump Casings	None	N/A	N/A	N/A
B-L-2	Pump Casings	None	N/A	N/A	N/A
B-M-1	Pressure Retaining Welds in Valve Bodies	2 Welds	1 Weld	50%	Yes
B-M-2	Valve Body > 4 in. Nominal Pipe Size	7 Valves	5 Valves	71.42%	Yes
B-N-1	Interior of Reactor Vessel	3 Items	2 Items	66.66%	No
B-N-2	Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels	None	N/A	N/A	N/A
B-N-3	Removable Core Support Structures	1 Item	None	0%	Yes
B-O	Pressure Retaining Welds in Control Rod Housings	3 Housings	3 Housings	100%	Yes
B-P	All Pressure Retaining Components				No
	System Leakage Test	160 Components	120 Components	75%	
	System Hydrostatic Test	20 Components	0 Components	0%	

Class 1 Inspections (Continued)

Section XI Category	Description	Inspections Required	Inspections Completed	Percentage Completed	Deferral Allowed
3-Q	Steam Generator Tubing	100% of Technical	Specifications Met		N/A
51.01	Class 1 Component Supports	280 Supports	214 Supports	76.42%	No

Class 2 Inspections

<u>Section XI Category</u>	<u>Description</u>	<u>Inspections Required</u>	<u>Inspections Completed</u>	<u>Percentage Completed</u>	<u>Deferral Allowed</u>
C-A	Pressure Retaining Welds In Pressure Vessels	9 Welds	6 Welds	66.66%	No
C-B	Pressure Retaining Nozzle Welds in Vessels	6 Welds	4 Welds	66.66%	No
C-C	Integral Attachments for Vessels, Piping, Pumps and Valves	33 Attachments	25 Attachments	75.75%	No
C-D	Pressure Retaining Bolting Exceeding 2 Inches in Diameter	None	N/A	N/A	N/A
C-F	Pressure Retaining Welds in Piping	343 Welds	271 Welds	79%	No
C	Pressure Retaining Welds in Pumps and Valves	25 Welds	15 Welds	60%	No
C-H	All Pressure Retaining Components				No
	System or Component Functional Test	70 Components	69 Components	98.57%	
	System Hydrostatic Test	90 Components	0 Components	0%	
F1.02	Class 2 Component Supports	568 Supports	437 Supports	76.93%	No

Augmented Inspections

<u>Description</u>	<u>Percentage Complete</u>
Reactor Coolant Pump Flywheels:	100% of Technical Specifications met
Steam Generator Tubes on Preheater Section	100% of requirements for Outages 1, 2, 3, 4, 5 & 6.
Main Steam Pipe Rupture Protection	100% of requirements for Outages 1, 2, 3, 4, 5 & 6.
Containment Spray System	100% of requirements for Outages 1, 2, 3, 4, 5 & 6.
Thermal Stress Piping	100% of requirements for Outage 3, 4, 5 & 6.
Safety Injection Accumulator Nozzles	100% of requirement for Outages 5 & 6.

3.0 Final Inservice Inspection Plan for Outage 6

The final ISI plan presented in this section lists all examinations credited for Outage 6 at Catawba 1. This includes ASME Section XI Class 1, 2 and augmented inspections required by the plant technical specifications.

The information shown below is a field description for the reporting format included in this section of the report:

Item Number	= ASME Section XI Tables IWB-2500-1 (Class 1), IWC-2500-1 (Class 2), IWD-2500-1 (Class 3), IWF-2500-1 (Class 1, Class 2 and Class 3), and Augmented Requirements
ID Number	= Unique Identification Number
Drawing Number	= Location and/or Detail Drawing
Locs.	= Location
Insp. Req.	= Examination Technique - Magnetic Particle, Dye Penetrant, etc.
Proc. Numbers	= Examination Procedures
Material Type/Grade	= General Description of Material
Diam./Thick	= Diameter/Thickness
Calib. Block	= Calibration Block
Comments	= General and/or Detail Description

5.8 Valves

Twelve (12) Studs and Nuts on Valves INC-1, INC-2 and INC-3 received a visual examination. No reportable conditions were found.

Valve body weld on Valve INC-27 received an ultrasonic examination. No reportable indications were found.

The internal surface of valves INC-27 and INI-129 received a visual examination. No reportable conditions were found.

5.9 System Leakage Test

The Class 1 Pressure Boundary was subjected to a system leakage test as required by ASME Section XI, Article IWB-5000. No reportable conditions were found.

5.10 Class 1 Repairs and Replacements

Repairs and replacements for work performed from June 10, 1991 to October 15, 1992 are itemized in Section 11 of this report.

6.0 Class 2 Inspection Results

Inspections were performed during Outage 6 on RHR Heat Exchanger 1A, Steam Generators 1B and 1D, piping integrally-welded attachments on Main Feedwater (CF) System and on Piping Welds of Auxiliary Feedwater (CA), Main Steam Supply to Auxiliary Equipment (SA), Main Steam (SM), Main Steam Vent to Atmosphere (SV), Main Feedwater (CF) and Safety Injection (SI) Systems. Examinations were also performed on Component Supports.

6.1 Residual Heat Removal Heat Exchanger and Steam Generators

Residual Heat Removal Heat Exchanger 1A Inlet Nozzle A and Reinforcing Pads, Outlet Nozzle B and Reinforcing Pads to Shell Welds received a dye penetrant examination per Problem Investigation report #O-C91-0439. No reportable indications were found.

Steam Generator 1B Main Steam Nozzle to Upper Head Weld and Steam Generator 1D Auxiliary Feedwater Nozzle to Shell Weld received a magnetic particle examination per Problem Investigation Report #O-C91-0439. No reportable indications were found.

A copy of Problem Investigation Report #O-C91-0439 is included in Section 10 of this report.

6.2 Piping

Welds one-half inch and less nominal wall thickness:

Twenty-six (26) circumferential welds and one (1) longitudinal weld received a magnetic particle examination. No reportable indications were found.

Welds greater than one-half inch nominal wall thickness:

Twelve (12) circumferential welds received an ultrasonic and a magnetic particle examination. Thirteen (13) circumferential welds received an ultrasonic and a dye penetrant examination. Three (3) circumferential welds received a radiographic and a magnetic particle examination. No reportable indications were found.

Piping Integrally-Welded Attachments:

Three (3) integrally-welded attachments on Main Feedwater (CF) System received a magnetic particle examination. No reportable indications were found.

6.3 Component Supports

Sixty-one (61) Class 2 Component Supports received visual examination as required by ASME Section XI, Article IWF-2000. No reportable conditions were found.

4.0 Results of Inspections Performed During Outage 6

The results of each inspection shown in the final ISI Plan (Section 3 of this report) are included in this section. The completion date and status for each inspection are shown. All inspections revealing reportable indications are described in further detail in Section 5, 6, or 7 as applicable.

The information shown below is a field description for the reporting format included in this section of the report:

Item Number = ASME Section XI Tables IWB-2500-1 (Class 1), IWC-2500-1 (Class 2), IWD-2500-1 (Class 3), IWF-2500-1 (Class 1, Class 2 and Class 3), and Augmented Requirements

ID Number = Unique Identification Number

Inspection Date = Date of Examination

Inspection Status

CLR = Clear
REC = Recordable
REP = Reportable

Inspection Limited

L = Limited
- = No

Geo. Ref. = Geometric Reflector

N = No
Y = Yes

Comments = General and/or Detail Description

PROGRAM: NISIRL JAISI04
FILE: C007133
PLANT: CATAMBA UNIT 1
KEY: ITEM NUMBER B02

DUKE POWER COMPANY
QUALITY ASSURANCE DEPARTMENT
PRE-SERVICE AND IN-SERVICE INSPECTION SYSTEM
CATAMBA 1 INSERVICE INSPECTION RESULTS OUTAGE 6

PAGE 1
DATE 12/18/92

ITEM NUMBER =====	ID NUMBER =====	INSPECTION DATE =====	INSPECTION STATUS =====	INSPECTION LIMITED =====	GEN. REF. =====	COMMENTS =====
B02.040.001	ISCA-01-02	07/16/92	CLR	L	Y	_____

6.4 System or Component Functional Test

Class 2 Functional Tests were performed as required by ASME Section XI, Article IWC-5000. No reportable conditions were found.

6.5 Class 2 Repairs and Replacements

Repairs and replacements for work performed from June 10, 1991 to October 15, 1992 are itemized in Section 11 of this report.

7.0 Augmented Inspection Results

Augmented inspections were performed on Reactor Coolant Pump Flywheels, Steam Generator Preheater Sections, Main Steam System for Pipe Rupture Protection and on Thermal Stress Piping (NRC Bulletin 88-08).

7.1 Reactor Coolant Pump Flywheels

Reactor Coolant Pump 1A Flywheel received an ultrasonic examination. Reactor Coolant Pump 1B Flywheel received an ultrasonic and a magnetic particle examination. No reportable indications were found.

7.2 Steam Generator Tube Preheater Section

All four steam generators were examined by eddy current methods. Results and evaluation data for examinations are shown in the Catawba Unit 1 July 1992 Refueling Outage Eddy Current Examination Report.

7.3 Pipe Rupture Protection

Eleven (11) welds on the Main Steam (SM) System received an ultrasonic and a magnetic particle examination. No reportable indications were found.

7.4 Thermal Stress Piping

Nine (9) welds and seven (7) bends were examined by ultrasonics on Reactor Coolant (NC) System, as required by USNRC Bulletin 88-08. No reportable indications were found.

9.0 Problem Investigation Reports

A copy of each Problem Investigation Report resulting from reportable items, originated against scheduled inservice inspections performed during Outage 6, is included in this section. All were resolved and found acceptable before returning Unit 1 to service. The following Problem Investigation Reports were issued.

<u>P.I.R NO.</u>	<u>Description</u>	<u>Date Issued</u>
None		

10.0 Reference Documents

The following reference documents apply to the inservice inspection performed during Outage 6 at Catawba 1:

- Duke Power Company Problem Investigation Report O-C91-0439
- Duke Power Company Technical Documentation #90-1 dated March 6, 1990

PIR SERIAL NO.: 0-C91-0439
 IIR SERIAL NO.: C91-087-0
 LER SERIAL NO.: 413/91-29 *Ken*
Station Report

CATAWBA NUCLEAR STATION
 PROBLEM INVESTIGATION PROCESS
 PROBLEM IDENTIFICATION FORM

I. Problem Investigation: (See Attached)

Occurred-Time/Date: _____ Discovered-Time/Date: _____

Unit(s) _____ Unit(s) Status at Time Problem Occurred: _____

Location of Problem: _____

Description and Cause of the Problem: _____

Immediate Corrective Actions Taken/To Be Taken: _____

Other Unit(s) Affected: () Yes () No

Problem Identified: Date _____ Section _____
 by _____

II. Significance: (See Attached)

Event Code _____ Initial Cause Code(s) 1. _____ 3. _____ Serial Number _____
 2. _____ 4. _____

Is this problem significant? () Yes () No () Delete

Responsible Section(s) 1. _____ 2. _____ 3. _____ 4. _____

Screened By: _____ Date: _____

Remarks: _____

DISTRIBUTION:

INITIAL	ORIGINATOR	JWGlen	RCFutrell	11-30-91 KE Nicholas	WRMcGillum	EMGeddie	JJForbes
WLC26C LEFrederick			CSRG	NRC	EGLaCasse	TPHarrall	JBFrye
FINAL	ORIGINATOR						

C91-081-0
~~443/91-29~~
Station Report

CATAWBA NUCLEAR STATION
PROBLEM INVESTIGATION PROCESS
PROBLEM IDENTIFICATION FORM

Problem ID

Occurred Time/Date UNKN / / Disc Time/Date 09/30/91

Unit(s) 0 Status at Time Occurred U1M1,U2EOC4

Location of Problem CNS

Description & Cause:

As a result of ONS PIR 4-091-0093, a data review of the Catawba Nozzle
to Vessel Welds revealed that the following item numbers were
inspected using the incorrect acceptance standard: Unit 1
CO2.011.001, CO2.011.002, CO2.021.004A, and CO2.021.012A, Unit 2
CO2.011-001, CO2.011.002 and CO2.021.051A

Immediate Corrective Actions:

The correct Unit 2 inspections are being done at 2EOC4

Problem Identified By: RNGILES Date 11/18/91 Section NON

Are Other Units Affected? (Y/N) N

I. Significance

Event Code

Initial Cause Code(s): A1b D1

Serial Number 91NON0177

Responsible Section(s): QA

Screened By: GBROBINS Date 11/19/91

Is the problem significant? Y

COMPLIANCE EVALUATION FORM

PROBLEM INVESTIGATION REPORT SERIAL NO.: 0-C91-0439

LICENSE EVENT REPORT NO.: 413/91-09 Kern
C91-087-0
Station Report

III. COMPLIANCE EVALUATION:

Operability Review By: J.W. Libby Date: 11-19-91

System/Component Operable: ☒ Yes ☐ No ☐ Not Applicable

Comments: Examining performed to Section XI standards revealed no defects.

Reportability Determination By: J.W. Libby Date: 11-19-91

Problem Reportable: ☒ Yes ☐ No

Reportable Per: 10 CFR 50.73 (a)(2)(i)(B) NA 12-18-91 940 per 50.73

Comments: Performance of the operation is a surveillance as defined by 10 CFR 50.73. Since the group
acceptance criteria was not met, the operator is required to meet the surveillance requirement
(see 50.73) and therefore continues operation prohibited by 50.73.

Notification Made (If Reportable to NRC) Time/Date: NA 1 NA

NRC Contactee: NA DPC Contractor: NA

Date Notified: NRC Res. Inspector: NA Station Manager: NA

Nuclear Support Duty Engineer: NA

Investigation Assigned To: C3RG (Station Report) NRC Report Due Date: 12-19-91 Kern

Date Due to Compliance After Evaluation: 12/12-91

Reportable Event PIR Review: RKS R. L. Fittrell Date: 12/18/91
Compliance Manager

Upper Tier Evaluation Assigned To: NA

Comments: NA

December 18, 1991

MEMO TO FILE

SUBJECT: Catawba Nuclear Station
Station Report
IIR C91-087-0; PIR 0-C91-0439
File No: CN-815.04

The attached report is being submitted as a Station Report concerning USE OF INCORRECT ACCEPTANCE STANDARDS DURING WELD INSPECTIONS DUE TO DEFECTIVE PROCEDURES.

W. R. McCollum, Station Manager
Catawba Nuclear Station

ken:STATION.RPT

xc: G. B. Robinson
J. W. Glenn
S. T. Rose
IIR C91-087-0

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
PROBLEM INVESTIGATION REPORT NO: 0-C91-0439

USE OF INCORRECT ACCEPTANCE STANDARDS DURING WELD
INSPECTIONS DUE TO DEFECTIVE PROCEDURES

ABSTRACT

On November 18, 1991, Unit 1 was operating in Mode 1, Power Operation, and Unit 2 was in refueling outage 2EOC4. A station Quality Assurance (QA) Specialist was reviewing weld inspection data, due to problems discovered with weld data at the Oconee Nuclear Station, when it was determined that the wrong acceptance standards had also been applied to the Catawba inspections. Under Technical Specification 3/4.0, Limiting Conditions For Operation And Surveillance Requirements, in-service inspection and testing of ASME Code Class 1, 2, and 3 equipment shall be performed in accordance with the ASME code, or as specified by Nuclear Regulatory Commission requirements. Inadvertently, the procedures directing the weld exams allowed application of less stringent acceptance criteria. A review of weld inspection data revealed that no flaw indications were present on the subject welds. Regardless of the applied acceptance criteria, inspectors record any relevant flaw indications on the weld inspection data sheets. This event has been attributed to Defective Procedures. The applicable Unit 2 welds have been re-inspected with satisfactory results. The remaining Unit 1 welds will be re-inspected during 1EOC6. Appropriate inspection procedures were revised with future revisions pending. Training on the procedure changes will be provided for the appropriate inspectors. This report is being submitted as a Station Report.

BACKGROUND

Technical Specification (T/S) 4.0.5 requires in-service inspection of ASME Code Class 1, 2, and 3 components and in-service testing of ASME Code Class 1, 2, and 3 pumps and valves to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and applicable Addenda, as required by 10CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission.

Per the Duke Power ASME Section XI Manual, Quality Assurance (QA) is responsible for weld inspection activities. All inspections shall meet the requirements of a ASME Section XI specified or as allowed in specific Code Cases or Relief Requests. Catawba's In-service Inspection Plan utilizes the 1980 Edition of ASME Code Section XI, with Addenda through Winter 1981.

Examination Categories for nozzle to vessel welds are given in Table IWB-2500-1 of Section XI, Article IWB-2000. At least 16% of the nozzle to vessel welds shall be examined by the end of the first inspection period (first 3 years of the first 10 years of operation), and a minimum of 50% at the end of the second period (years 4-7).

The General Office (GO) QA ISI Specialist who oversees Catawba's ISI Program uses the information obtained from ASME Section XI, and other sources, and works with the QA Catawba Site ISI Coordinator to establish the timetables for all welds that require in-service inspection at Catawba.

The examination techniques used for in-service inspections include Radiographic (RT), Ultrasonic (UT), Magnetic Particle (MT), Liquid Penetrant (PT), Eddy Current, and Visual examination methods. For all examinations, both remote and manual, specific procedures are prepared describing the equipment, inspection technique, operator qualifications, calibration standards, flaw evaluation, and records. The In-service Inspection Plan for Unit 2 was submitted on August 18, 1986. The in-service inspection interval is every 10 years (4 total). Detailed inspection listings and scheduling are contained within the Catawba In-service Inspection Plan.

The Non-Destructive Examination procedures (NDE-25 and NDE-35) states " For Class 1 or Class 2 components where acceptance standards are not specified, one of the following acceptance standards shall be used:

- (1) ASME Section III edition applicable to the construction.
- (2) ASME Section XI Class 1.
- (3) The original construction code (excludes NDE-35).

The inspections involved with this incident are; Residual Heat Removal Heat Exchanger (Hx) 1A/2A Inlet and Outlet Nozzles; Steam Generator 1B Upper Main Steam Nozzle; Steam Generator 1D Auxiliary Feedwater Nozzle To Shell; and the Safety Injection Accumulator Tank 2B Outlet Nozzle.

EVENT DESCRIPTION

On September 9, 1991, an Oconee Nuclear Station (ONS) QA Specialist assigned to the General Office was reviewing ISI data when it was discovered that a flaw indication recorded during an inspection was initially determined to be acceptable. The dimensions of the flaw were recorded on the inspection data sheet were compared to the inspection acceptance criteria. It was determined that the flaw was actually unacceptable. It was concluded that the wrong acceptance criteria had been applied to the inspections that had been performed. ONS Problem Investigation Report (PIR) 4-091-0093 was initiated.

On November 18, 1991, Catawba QA performed a review of their ISI data recorded for the nozzle to vessel welds in response to the ONS PIR. This review revealed that the CNS inspections also employed the wrong acceptance criteria.

The three Unit 2 welds were re-inspected using the correct acceptance criteria between November 19th and 20th. No flaw indications exceeded the acceptable limits of the inspections.

The Unit 1 ND Hx welds were inspected on November 27, 1991. No unacceptable indications were discovered. The Unit 1 S/G 1B upper main steam nozzle and S/G 1D CA weld inspections have been scheduled for 1EOC6.

CONCLUSION

This incident is attributed to Defective Procedures because the procedures listed multiple acceptance criteria for the inspections.

The initial nozzle to vessel weld inspections were performed using the wrong acceptance criteria because QA selected ASME Section XI acceptance criteria for the inspections instead of the more stringent construction code standards.

A subsequent review of the weld inspection data sheets revealed that no flaw indications were observed by the inspectors. Weld inspectors record any flaw indications ($> 1/16"$) regardless of the acceptance criteria limits applied to the inspection. Therefore, it was determined that the welds inspected under the subject circumstances are acceptable under the criteria established from the construction code.

Due to the problem discovered at ONS, NDE-25 and NDE-35 were revised on October 15, 1991, to require use of Appendix A (section 16.5.1 of NDE-25 and 13.4.1 of NDE-35) to specify acceptance criteria when acceptance standards are not specified for Class I and II components. Further revisions to NDE-25 and NDE-35 will add the requirement for documentation of any relevant indications regardless that the indication does not exceed the applied acceptance criteria.

CORRECTIVE ACTION

SUBSEQUENT

- 1) The affected Unit 2 nozzle to vessel welds were re-inspected using the proper acceptance criteria.
- 2) Unit 1 RHR Hx 1A nozzle to vessel welds were re-inspected using the proper acceptance criteria.

PLANNED

- 1) NDE-25 and NDE-35 will be revised to require that all relevant indications exceeding 1/16" will be recorded .
- 2) QA inspectors will be trained on selecting acceptance criteria for components that do not have specified standards.
- 3) Unit 1 S/G 1B Main Steam nozzle to upper head weld will be re-inspected using the proper acceptance criteria.
- 4) Unit 1 S/G 1D Auxiliary Feedwater nozzle to shell weld will be re-inspected using the proper acceptance criteria.

SAFETY ANALYSIS

No flaw indications were detected during subsequent reinspections using the correct acceptance criteria. No adverse impact was inflicted upon the performance or operability of the systems affected by this incident.

The health and safety of the public were not affected by this incident.

TECHNICAL DOCUMENT

NO. 90-1

SUBJECT:

AN ALTERNATIVE ULTRASONIC EXAMINATION
FOR
THE MAIN STEAM HEADER TRANSITION PIECE

DUKE POWER APPROVAL SIGNATURES

QA MANAGER-ISI: *[Signature]* DATE 3/6/90

LEVEL III UT: *[Signature]* DATE 3/5/90

ANII CATAWBA: *[Signature]* DATE 3-6-90

ANII MCGUIRE: *[Signature]* DATE 3-5-90

1.0 PURPOSE:

This paper pertains to the qualification of an alternative ultrasonic technique that will be employed at both McGuire and Catawba Nuclear Power Stations. The component is the main steam system. The weld joint of interest is the main steam header (1) to transition piece (2) weld (see figure # 1).

In accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME), an alternate examination technique may be substituted for techniques specified in the Code, provided that the Authorized Nuclear Inservice Inspector is satisfied that the results are demonstrated to be equivalent or superior to those of the specified method (see paragraphs III-1100 (d) and IWA-2240 for more detail).

2.0 BACKGROUND

Inservice examinations are conducted in accordance with the 1980 edition of ASME Section XI, including all addenda through Winter 1980 at the McGuire Nuclear Station. At the Catawba Nuclear Station, all inservice examinations are conducted in accordance with the the 1980 edition of ASME Section XI, including all addenda through Winter 1981. Appendix III, in both editions of above mentioned Codes, stipulates that an ultrasonic examination calibration block, for Class 1 and 2 ferritic piping, be of the same nominal diameter as the piping component to be examined.

Due to the unique design of the transition piece, locating representative material to produce a calibration standard of the same dimensions has not been possible to date. The transition piece tapers from a 10" outside dimension (O.D.), on the main steam header side, to 9.0" O.D. on the valve side. Ninety-six welds are of this configuration (forty-eight at each unit). The header side weld's O.D. varies from 11.5" to 12.0".

3.0 SUPPOSITION:

The valve side of the transition piece calibration is established with a 9.0" O.D. calibration block. The supposition is that this calibration block will provide a examination sensitivity equivalent or superior to one that would be obtained from a calibration block designed to the specifications of the Code. This supposition is based on the following:

- 3.1 Due to the geometric configuration of the header to transition piece weld joint, the examination shall only be performed from the transition side of the weld.
- 3.2 The thickness of the 9.0" O.D. calibration block and the header side of the transition piece are the same. Therefore, the calibration block reflector size and depth location requirements would be the same for a 12.0" O.D. calibration block as are in the 9.0" O.D. block.
- 3.3 The surface area of the 9.0" O.D. block would tend to reflect more of the sound transmitted from a given transducer than the surface area of a 12.0" O.D. block. This would allow less sound to reach the reflective surface of the calibration block reflectors. This means that more gain would be required to obtain the same amplitude from the calibration reflectors in the 9.0" O.D. calibration block than 12.0" block. Therefore a calibration from the 9.0" O.D. block should provide a more sensitive examination.

4.0 QUALIFICATION PROCESS

The 9.0" O.D. calibration blocks from both the McGuire Nuclear Station (50249) and Catawba Nuclear Station (50356) were compared to a 12" O.D. calibration block (50252). All the calibration blocks are of the same thickness and contain the same

QUALIFICATION PROCESS (CONT.)

size calibration reflectors. The only difference is block 50252 does not have a notch reflector. The same instrumentation (UT instrument, transducer, and cable) were utilized throughout the qualification process. Without changing the UT instrument settings, calibrations were constructed in accordance with the Babcock and Wilcox piping examination procedure ISI-120, revision 25. Each calibration was documented and any deviations were noted (see Calibration Sheets 90-1A through 90-1F).

The comparisons were accomplished using a certified 2.25.MHz, .50" diameter, transducer (32908) coupled to a 6' microdot cable. The ultrasonic instrument was a Tokyo Keiki, model SM-100. No thermometer was used. The comparisons were performed in a environmentally controlled location. Each of the calibration blocks was in this environment 24 hours prior to the start of the comparisons and remained there throughout the process.

5.0 RESULTS:

The initial distance amplitude correction (DAC) curve was performed on the 9.0" O.D. calibration block 50356 using the side drilled holes. The I.D. notch was compared to this DAC curve. The notch is use to determine the calibration curve's sensitivity to I.D. surface initiated degradation. The DAC curve was 10 Decibels (dB) more sensitive than the notch at reference level. This means that the DAC curve's sensitivity is more than 3 times greater than the calibration sensitivity obtained from the notch. Without changing the sensitivity of the instrument, a calibration was established from the side drilled holes in the 12" O.D. block, 50252. The amplitude obtained from these reflectors was less than 2 dB greater than the reference level of block 50356 (The SM-100 adds and subtracts gain sensitivity in 2 dB increments). Block 50252 does not have a notch, so no notch comparison was made.

5.0 RESULTS (CONT.):

A calibration was then generated from the other 9.0" O.D. block, 50249. The DAC curve obtained was identical to the one obtained from the 12" O.D. block, 50252. It was not possible to measure the difference in sensitivity between these two calibration curves with the instrument control settings. A graph of all 3 DAC curves depicts these results in Figure No. 2. The I.D. notch was then compared to this DAC curve. The DAC curve was again 10 dB more sensitive than the notch at reference level.

DAC curves were generated on each calibration block in the axial and circumferential direction. In each case the calibration obtained from the DAC curve was at least 10 dB greater in sensitivity than the sensitivity obtained from the notch.

6.0 CONCLUSION :

The 9.0" O.D. calibration block can be used for the examination of both welds attached to the transition piece on main steam at both the Catawba and McGuire Nuclear Stations. The difference in examination sensitivity using any of the three blocks was < 2 dB. A 2 dB change is recognized by Code and throughout the industry to be an insignificant change in a calibration sensitivity. In any case, calibration blocks 50356 and 50249 provided an examination sensitivity equal to or greater than that obtained from the 12" O.D. calibration block, 50252.

Using a calibration block that has no notch reflector for this comparison did not hinder the results. A DAC calibration obtained from side drill holes normally results in a more sensitive calibration than one obtained from a notch. In all cases where a notch was available the DAC calibration sensitivity was 10 dB more sensitive than that obtained from the notch reflectors.

6.0 CONCLUSION (CONT.):

This comparison was performed with a .50 diameter 2.25 MHz transducer. Since no other transducers were demonstrated, this alternative examination technique shall only be performed with the transducer frequency and size demonstrated.

This comparison shall not be used as a justification for any other alternative examination technique other than the main steam header to transition piece valve weld.

- (1) MAIN STEAM HEADER
 - (2) 10"X9" TRANSITION PIECE
 - (3) NOZZLE TYPE SAFETY VALVE
(POP-OFF VALVE)
- CROSBY REFERENCE DWG'S
SA-G-K56314 (10" OUTLET)
SA-G-K56315 (8" OUTLET)

FOR INFORMATION ONLY

FIGURE # 1

Babcock & Wilcox

a McDermott company

CALIBRATION SHEETDATE: Feb 1/90 , TIME: 8:00 HRS

CUSTOMER: DUKE POWER CO. CONTRACT NO.: N/A COMPONENT: PIPING PROCEDURE: 120 R.25
 EXAMINER: R. J. PAINE I.D. NO.: P1680 LEVEL: II COUPLANT: ULTRAGELII ID/BATCH: 9981
 EXAMINER: N/A I.D. NO.: N/A LEVEL: N/A THERMOMETER I.D.: ☼ CAL DUE: ☼

INSTRUMENT CALIBRATION DATA:

INSTRUMENT I.D. NO.: 1348
 CAL. DUE: 3-14-90 LINEARITY CHECK: ☐ YES ☒ NO
 INST. TYPE: ☐ USK ☐ EPOCH(2002) ☒ SM-100
 REJECT: ☒ OFF ☐ ON, %FSH
 MAT. CAL./VEL.(2002): 355
 DELAY: 891
 PULSE ENERGY: 2 ZERO(2002): N/A
 DAMPENING(2002): N/A OHM. X % N/A
 GAIN(COARSE): 20 Db. 2Db STEP GAIN: 12 Db
 CONTINUOUS FINE GAIN: 0 %, 0 Db
 SCREEN RANGE: 125 SCREEN DEPTH: 2.5 "
 OPERATION: ☐ TBR (2002): ☐ THRU
☒ NORMAL ☐ TBR
 DISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE +
N ☐ HALF WAVE -
 FREQ./FILTER(2002): ☐ BROADBAND A
☐ MHZ
 MEMORY(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK

CRYSTAL
 I.D. NO.: 50356 ID. NO.: 32908
 LENGTH: 12.0 " ☐ TBR ☒ LWS ☐ LW
 O.D.: 9.0 " FREQ.: 2.25 MHZ
 THICKNESS: 1.5 " SIZE: .5 "
 TEMP.: ☼ °F ACTUAL X 44 DEG.

SYSTEM CALIBRATIONANGLE: 45 ° CAL. DIR.: ☒ AXIAL ☐ CIRC.

REFLECTOR	AMPLITUDE %FSH	SCREEN READING IN./MM
1 /B NODE	80 %	.40 "
2 /B NODE	79 %	.80 "
3 /B NODE	68 %	1.15 "
5 /B NODE	29 %	1.85 "
/B NODE	%	
/B NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

CALIBRATION BLOCK SIMULATOR

SERIAL NO.: N 2Db STEP GAIN: Db
 SCREEN RANGE: N TEMP.: °F
 SIGNAL DEPTH:
 SIGNAL AMP: %FSH A
 COARSE GAIN: Db

SEARCH UNIT CABLECABLE TYPE: LENGTH: MICRODOT 6 "**NOTES:**

NOTCH GAIN = D.A.C. at 20 + 2db
 at 1.5 " Depth
 ☼ Thermometer not used, all blocks
 were at the same location for
 previous 24 hrs. and are therefore
 at the same ambient temperature.

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SIM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
/B NODE	%	%	%	%
/B NODE	%	%	%	%
/B NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOTCH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:N/A

REVIEWED BY: [Signature] , LEVEL: II , DATE: 2-1-90 CAL. SHEET NO. 90-1A
 REVIEWED BY: TIT F: DATE:

Babcock & Wilcox

a McDermott company

CALIBRATION SHEETDATE: Feb 1/90 , TIME: 8:06 HRSCUSTOMER: DUKE POWER CO. CONTRACT NO.: N/A COMPONENT: PIPING PROCEDURE: 120 R.25EXAMINER: R. J. PAINE I.D. NO.: P1680 LEVEL: II COUPLANT: ULTRAGEL II ID/BATCH: 8981EXAMINER: N/A I.D. NO.: N/A LEVEL: N/A THERMOMETER I.D.: -/- CAL DUE: -/-**INSTRUMENT CALIBRATION DATA:**INSTRUMENT I.D. NO.: 1345CAL DUE: 3-13-90 LINEARITY CHECK: ☐ YES, ☒ NOINST. TYPE: ☒ USR FB 1/1, ☐ EPOCH(2002), ☒ SM-100REJECT: ☒ OFF, ☐ ON, %FSHMAT. CAL./VEL.(2002): 353DELAY: 904PULSE ENERGY: 2 ZERO(2002): N/ADAMPENING(2002): N/A OHM. 4 N/AGAIN(COARSE): 20 Db. 2Db STEP GAIN: 16 DbCONTINUOUS FINE GAIN: 0 %, 0 DbSCREEN RANGE: 125 mm SCREEN DEPTH: 2.5 "OPERATION: ☐ TBR (2002): ☐ THRU
☒ NORMAL ☐ TBRDISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE +
N ☐ HALF WAVE -FREQ./FILTER(2002): ☐ BROADBAND A
☐ MHZ

MEMORY(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK**CRYSTAL**I.D. NO.: 50356 ID. NO.: 32908
LENGTH: 12.0 " ☐ TBR ☒ LS ☐ LW
O.D.: 9.0 " FREQ.: 2.25 MHZ
THICKNESS: 1.5 " SIZE: .5 "
TEMP.: -/- °F ACTUAL 44 DEG.**CALIBRATION BLOCK SIMULATOR**SERIAL NO.: N 2Db STEP GAIN: Db
SCREEN RANGE: N TEMP.: °F
SIGNAL DEPTH:
SIGNAL AMP: %FSH A
COARSE GAIN: Db**SYSTEM CALIBRATION**ANGLE: 45 °; CAL. DIR.: ☐ AXIAL, ☒ CIRC.

REFLECTOR	AMPLITUDE %FSH	SCREEN READING IN/MM
1 /8 NODE	80 %	.40 "
2 /8 NODE	65 %	.85 "
3 /8 NODE	48 %	1.3 "
5 /8 NODE	* %	* "
/8 NODE	%	
/8 NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

SEARCH UNIT CABLECABLE TYPE: MICRODOT LENGTH: 6 '**NOTES:**

NOTCH = D.A.C. at 20+4 db
at 2.95" Depth.
-/- Thermometer not used all
blocks were at the same location
for previous 24hrs. and are
therefore at the same ambient
temperature.
* Due to curvature unable
to obtain 5/8 node.

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SIM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOTCH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:N/AREVIEWED BY: [Signature] LEVEL: II DATE: 2-1-90
RFVIEWED BY: TITLE: DATE: CAL. SHEET NO. 90-1BPAGE 2 OF 6

Babcock & Wilcox
a McDermott company

CALIBRATION SHEET

DATE: Feb 1/90, TIME: 8:39 HRS

CUSTOMER: <u>DUKE POWER CO.</u>	CONTRACT NO.: <u>N/A</u>	COMPONENT: <u>PIPING</u>	PROCEDURE: <u>120 R.25</u>
EXAMINER: <u>R.J. PAINE</u>	I.D.NO.: <u>P1680</u>	LEVEL: <u>II</u>	COUPLANT: <u>ULTRAGEL II</u>
EXAMINER: <u>N/A</u>	I.D.NO.: <u>N/A</u>	LEVEL: <u>N/A</u>	THERMOMETER I.D.: <u>-/-</u>
			CAL DUE: <u>-/-</u>

INSTRUMENT CALIBRATION DATA:

INSTRUMENT I.D. NO.: 1348

CAL DUE: 2-14-90 LINEARITY CHECK: ☐ YES, ☒ NO

INST. TYPE: ☐ USK, ☐ EPOCH(2002), ☒ SM-100

REJECT: ☒ OFF, ☐ ON, _____ % FSH

MAT. CAL./VEL.(2002): 355

DELAY: 891

PULSE ENERGY: 2 ZERO(2002): N/A

DAMPENING(2002): N/A OHM. x _____ % N/A

GAIN(COARSE): 20 Db. 2Db STEP GAIN: 12 Db

CONTINUOUS FINE GAIN: 0 %, 0 Db

SCREEN RANGE: 125mm SCREEN DEPTH: 2.5"

OPERATION: ☐ TBR (2002): ☐ THRU
☒ NORMAL ☐ TBR

DISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE +
☒ HALF WAVE -

FREQ./FILTER(2002): ☐ BROADBAND A
☐ _____ MHZ

MEMORY(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK CRYSTAL

I.D.NO.: 50252 I.D.NO.: 32908

LENGTH: 12.0" ☐ TBR ☒ LS ☐ LW

O.D.: 12" FREQ.: 2.25 MHZ

THICKNESS: 1.5" SIZE: .5"

TEMP.: -/- °F ACTUAL x 44 DEG.

CALIBRATION BLOCK SIMULATOR

SERIAL NO.: _____ 2Db STEP GAIN: _____ Db

SCREEN RANGE: N TEMP.: _____ °F

SIGNAL DEPTH: _____

SIGNAL AMP: _____ % FSH

COARSE GAIN: _____ Db

SYSTEM CALIBRATION

ANGLE: 45° CAL. DIR.: ☒ AXIAL, ☐ CIRC.

REFLECTOR	AMPLITUDE % FSH	SCREEN READING IN/MM
1 /8 NODE	94 %	.40"
2 /8 NODE	86 %	.80"
3 /8 NODE	70 %	1.20"
5 /8 NODE	36 %	1.90"
/8 NODE	%	
/8 NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

SEARCH UNIT CABLE

CABLE TYPE: MICRODOT LENGTH: 6'

NOTES:

No. notches in block.

-/- Thermometer not used, all blocks were at the same location for previous 24 hrs. and are therefore at the same ambient temperature.

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SIM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOTCH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:

N/A

REVIEWED BY: [Signature] LEVEL: II DATE: 2-1-90 CAL. SHEET NO. 90-10

DATE: _____

C-72

Babcock & Wilcox

a McDermott company

CALIBRATION SHEETDATE: Feb 1/90, TIME: 8:45 HRS

CUSTOMER: <u>Duke Power Co.</u>	CONTRACT NO.: <u>N/A</u>	COMPONENT: <u>PIPING</u>	PROCEDURE: <u>120 R.25</u>
EXAMINER: <u>R. J. PAINE</u>	I.D. NO.: <u>P1680</u>	LEVEL: <u>II</u>	COUPLANT: <u>ULTRAGEL II</u>
EXAMINER: <u>N/A</u>	I.D. NO.: <u>N/A</u>	LEVEL: <u>N/A</u>	THERMOMETER I.D.: _____
		CAL DUE: _____	

INSTRUMENT CALIBRATION DATA:

INSTRUMENT I.D. NO.: 1345

CAL DUE: 3-13-90 LINEARITY CHECK: ☐ YES, ☒ NO

INST. TYPE: ☐ USK, ☐ EPOCH(2002), ☒ SM-100

REJECT: ☒ OFF, ☐ ON, _____ % FSH

MAT. CAL./VEL.(2002): 353

DELAY: 904

PULSE ENERGY: 2 ZERO(2002): N/A

DAMPENING(2002): N/A OHM. X _____ % N/A

GAIN(COARSE): 20 Db, 20db STEP GAIN: 16 Db

CONTINUOUS FINE GAIN: 0 %, 0 Db

SCREEN RANGE: 125mm SCREEN DEPTH: 2.5"

OPERATION: ☐ TBR (2002): ☐ THRU
☒ NORMAL ☐ TBR

DISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE +
☒ HALF WAVE -

FREQ./FILTER(2002): ☐ BROADBAND A
☐ _____ MHZ

MEMORY(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK CRYSTAL

I.D. NO.: 50252 ID NO.: 32908

LENGTH: 12.0" ☐ TBR ☒ LWS ☐ LW

O.D.: 12" FREQ.: 2.25 MHZ

THICKNESS: 1.5" SIZE: 1.5"

TEMP.: -7 °F ACTUAL 44 DEG.

SYSTEM CALIBRATION

ANGLE: 45 ° CAL. DIR.: ☐ AXIAL, ☒ CIRC.

REFLECTOR	AMPLITUDE % FSH	SCREEN READING IN/MM
1 /8 NODE	90 %	4.0"
2 /8 NODE	80 %	1.5"
3 /8 NODE	58 %	1.3"
5 /8 NODE	22 %	2.1"
/8 NODE	%	
/8 NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

CALIBRATION BLOCK SIMULATOR

SERIAL NO.: _____ 20db STEP GAIN: _____ Db

SCREEN RANGE: A TEMP.: _____ °F

SIGNAL DEPTH: _____

SIGNAL AMP: _____ % FSH

COARSE GAIN: _____ Db

SEARCH UNIT CABLE

CABLE TYPE: MICRODOT LENGTH: 6'

NOTES:

No notches in block

Thermometers not used, all blocks were at the same location for previous 24 hrs. and are therefore at the same ambient temperature.

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SIM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
/8 NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOICH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:

N/A

REVIEWED BY: [Signature], LEVEL: II, DATE: 2-1-90 CAL. SHEET NO. 90-1D

REVIEWED BY: _____, TITLE: _____, DATE: _____

PAGE 4 OF 6

Babcock & Wilcox

a McDermott company

CALIBRATION SHEETDATE: Feb 1 / 90 TIME: 8:17 HRS

CUSTOMER: <u>DUKE POWER CO.</u>	CONTRACT NO.: <u>N/A</u>	COMPONENT: <u>PIPING</u>	PROCEDURE: <u>120 R25</u>
EXAMINER: <u>R. J. PAINE</u>	I.D. NO.: <u>P1680</u>	LEVEL: <u>II</u>	COUPLANT: <u>ULTRAGEL II</u>
EXAMINER: <u>N/A</u>	I.D. NO.: <u>N/A</u>	LEVEL: <u>N/A</u>	THERMOMETER I.D.: <u>-</u>
			CAL DUE: <u>-</u>

INSTRUMENT CALIBRATION DATA:

INSTRUMENT I.D. NO.: 1348

CAL. DUE: 3-14-90 LINEARITY CHECK: ☐ YES ☒ NO

INST. TYPE: ☐ USK ☐ EPOCH(2002) ☒ SM-100

REJECT: ☒ OFF ☐ ON, _____ %FSH

MAT. CAL./VEL.(2002): 355

DELAY: 891

PULSE ENERGY: 2 ZERO(2002): N/A

DAMPENING(2002): N/A OHM. 4 % N/A

GAIN(COARSE): 20 Db. 2Db STEP GAIN: 12 Db

CONTINUOUS FINE GAIN: 0 %, 0 Db.

SCREEN RANGE: 125 mm SCREEN DEPTH: 2.5"

OPERATION: ☐ TBR (2002): ☐ THRU ☒ NORMAL ☐ TBR

DISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE + ☒ HALF WAVE -

FREQ./FILTER(2002): ☐ BROADBAND ☒ _____ MHZ A

MEMOEX(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK CRYSTAL

I.D. NO.: 50249 I.D. NO.: 32908

LENGTH: 12.0" ☐ TBR ☒ LS ☐ LW

O.D.: 9.8" FREQ.: 2.25 MHZ

THICKNESS: 1.60" SIZE: .5"

TEMP.: - °F ACTUAL 44 DEG.

SYSTEM CALIBRATION

ANGLE: 45 ° CAL. DIR.: ☒ AXIAL ☐ CIRC.

REFLECTOR	AMPLITUDE %FSH	SCREEN READING IN/MM
1/8 NODE	95 %	.40"
2/8 NODE	96 %	.80"
3/8 NODE	70 %	1.20"
5/8 NODE	36 %	2.00"
1/8 NODE	%	
1/8 NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

CALIBRATION BLOCK SIMULATOR

SERIAL NO.: N/A 2Db STEP GAIN: _____ Db

SCREEN RANGE: N/A TEMP.: _____ °F

SIGNAL DEPTH: A

SIGNAL AMP: _____ %FSH

COARSE GAIN: _____ Db

SEARCH UNIT CABLE

CABLE TYPE: MICRODOT LENGTH: 6'

NOTES:

NOTCH = D.A.C. at 20 + 2db at 1.6" Depth

Thermometer not used, all blocks were at the same location for previous 24hrs. and are therefore at the same ambient temperature

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SIM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
1/8 NODE	%	%	%	%
1/8 NODE	%	%	%	%
1/8 NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOTCH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:

N/A

REVIEWED BY: [Signature] LEVEL: II DATE: 2-1-90 CAL. SHEET NO. 90-1E

Babcock & Wilcox

a McDermott company

CALIBRATION SHEETDATE: Feb 1/90 TIME: 8:28 HRSCUSTOMER: DUKE POWER CO. CONTRACT NO.: N/A COMPONENT: PIPING PROCEDURE: 120 R 25EXAMINER: R. J. PAINE I.D. NO.: P1680 LEVEL: II COUPLANT: UTR GEL II ID/BATCH: 8981EXAMINER: NP I.D. NO.: N/A LEVEL: N/A THERMOMETER I.D.: - CAL DUE: -**INSTRUMENT CALIBRATION DATA:**INSTRUMENT I.D. NO.: 1345CAL DUE: 3-13-90 LINEARITY CHECK: ☐ YES, ☒ NOINST. TYPE: ☐ USK ☐ EPOCH(2002), ☒ SM-100REJECT: ☒ OFF, ☐ ON, %FSHMAT. CAL./VEL.(2002): 353DELAY: 904PULSE ENERGY: 2 ZERO(2002): N/ADAMPENING(2002): N/A OHM. 4 1 N/AGAIN(COARSE): 20 Db. 2Db STEP GAIN: 16 DbCONTINUOUS FINE GAIN: 0 %, 0 DbSCREEN RANGE: 125mm SCREEN DEPTH: 2.5"OPERATION: ☐ TBR (2002): ☐ THRU
☒ NORMAL ☐ TBRDISPLAY(2002): ☐ FULL WAVE, ☐ HALF WAVE +
☒ HALF WAVE -FREQ./FILTER(2002): ☐ BROADBAND A
☐ MHZ

MEMORY(2002): 1 2 3 4 5 6 7 8 9 10

CALIBRATION BLOCK**CRYSTAL**I.D. NO.: 50249 I.D. NO.: 32908
LENGTH: 12.0" ☐ TBR ☒ KS ☐ LW
O.D.: 9.8" FREQ.: 2.25 MHZ
THICKNESS: 1.60" SIZE: .5"
TEMP.: - °F ACTUAL 44 DEG.**SYSTEM CALIBRATION**ANGLE: 45° CAL. DIR.: ☐ AXIAL, ☒ CIRC.

REFLECTOR	AMPLITUDE %FSH	SCREEN READING IN/MM
1/8 NODE	86 %	.50"
2/8 NODE	75 %	.95"
3/8 NODE	55 %	1.50"
5/8 NODE	* %	*
7/8 NODE	%	
7/8 NODE	%	
OTHER	%	
OPPOSITE NOTCH	%	
BKR CB	%	
BKR P	%	

CALIBRATION BLOCK SIMULATORSERIAL NO.: 2Db STEP GAIN: Db
SCREEN RANGE: N TEMP.: °F
SIGNAL DEPTH:
SIGNAL AMP: %FSH A
COARSE GAIN: Db**SEARCH UNIT CABLE**CABLE TYPE: LENGTH: MICRODOT 6'**NOTES:**

NOTCH = DAC at 20 db at
2.05" depth
Thermometer not used, all
blocks were at the same location
for previous 24 hrs. and are therefore
at the same ambient temperature.
* DUE TO CURVATURE UNABLE
TO OBTAIN 5/8 NODE.

CALIBRATION CONFIRMATION

TIME	HRS	HRS	HRS	HRS
CAL. BLOCK SM.	%	%	%	%
BACK REFL. CB/P	%	%	%	%
7/8 NODE	%	%	%	%
7/8 NODE	%	%	%	%
7/8 NODE	%	%	%	%
OTHER	%	%	%	%
OPPOSITE NOTCH	%	%	%	%
INITIALS				

FIGURE NO(S) EXAMINED:N/A
 REVIEWED BY: [Signature] LEVEL: II DATE: 2-1-90 CAL. SHEET NO. 90-1F
 REVIEWED BY: TITLE: DATE: PAGE 6 OF 6

CALIBRATION BLOCK CURVE COMPARISON

FIGURE # 2

11.0 Class 1 and 2 Repairs and Replacements

As required by ASME Section XI, 1980 Edition, a record of the Class 1 and 2 Repairs and Replacements for work performed from June 10, 1991 to October 15, 1992 is provided and is included in this section of the report. The individual work request documents are on file at the Catawba Nuclear Station.

CATAWBA NUCLEAR STATION
UNIT 1 EOC 6
REPAIR / REPLACEMENT LOG
ASME SECTION XI-1981

Examination Dates: From: 06/10/91 To: 10/15/92

Prepared By: QA Technical Support

Transmitted By: R.C. Allen Date: 10-22-92

The following is
easier to read
than a stack of
NIS-2 forms.

REPAIR/REPLACEMENT LOG
ASME SECTION XI 1981
U1EOC6

WR#	Class	Description
4193 MES	B	REPLACED SNUBBER 1-R-ND-0596
4281 MES	B	REPLACED LOAD PIN, 1-R-CF-1567
4918 MES	B	REPLACED SNUBBER BODY 1-R-ND-0226
4918 MES	B	REPLACED SNUBBER BODY 1-R-ND-0596
13716 NSM	B	REPLACED PLATE 1-R-ND-0332
13716 NSM	B	REPLACED STRUT, REAR BRACKET, PIPE CLAMP, TUBE STEEL 1-R-ND-0331
13730 NSM	B	REPLACED STRUT & PIPE CLAMP, 1-R-CA-1552
82028310-01	B	REPLACED TRUNION ASSEMBLY ON S.W.I. FILTER HOUSING
91014154-01	B	REPLACED BONNET STUDS VALVE 1SV27A
91014155-01	B	REPLACED BONNET STUDS VALVE 1SV28A
91036863-01	B	NV PUMP 1A (REPLACED MECH. SEALS)
91036868-01	B	VALVE 1NV294 REPLACED BONNET STUD NUTS
91061495-01	A	BROKE AND REMADE BONNET JOINT ON VALVE 1NI157
91061797-01	B	VALVE 1NV186A (REPLACED YOKE AND BONNET)
91079959-01		REPLACED VALVE 1ND-03 OLD SERIAL#TD89405 NEW SERIAL#TD89407
91079977-01	B	REPLACED VALVE 1NV-14
92003090-01	B	REPLACE VALVE 1NV291, 3"-CUT-OUT WELDS, 1NV43-12,13 (NSM11162)

REPAIR/REPLACEMENT LOG
ASME SECTION XI 1981
U1EOC6

92010120-01	A	REACTOR COLLANT PUMP 1C (REPLACED SEALS)
92014280-01	A	REPLACED VALVE 1NC-2
92014280-01	A	VALVE 1NC002 (REPLACED)
92014312-01	A	REMOVAL/REPLACED REACTOR VESSEL HEAD (REPLACED RV STUDS)
92014312-01	A	REMOVED THREADS - REACTOR VESSEL FLANGE
92014320-01	A	RX THERMO COUPLE ASSEMBLY (REMOVE AND REPLACE
92015141-01	A	DISASSEMBLED VALVE 1NI71 (NO PRESSURE BOUNDARY PARTS REPLACED)
92015253-01	A	DISASSEMBLED VALVE 1NI29 (NO PRESSURE BOUNDARY PARTS REPLACED)
92015254-01	A	DISASSEMBLED VALVE 1NI180 (NO PRESSURE BOUNDARY PARTS REPLACED)
92019184-01	A	REPLACED STEM ON VALVE 1NC31B
92019185-01	A	REPLACED STEM ON VALVE 1NC33A
92019186-01	A	REPLACED STEM ON VALVE 1NC35B
92019564-01	B	REPLACED ANCHORS, PIPE CLAMP, LEFT HAND ROD EYE, 1-R-SV-1620
92022273-01	A	VALVE 1NC003 (REPLACED)
92022274-01	A	DISASSEMBLED VALVE 1NI70 (NO PRESSURE BOUNDARY PARTS REPLACED)
92022459-01	A	REPLACED TWO SNUBBER 1-R-NI-1384
92022460-01	B	REPLACED SNUBBER, 1-R-SM-1578, & 1-R-CF-1519
92032839-01	A	DISASSEMBLE AND REASSEMBLE PRIMARY MANWAY SG1C

REPAIR/REPLACEMENT LOG
ASME SECTION XI 1981
U1EOC6

92032851-01	A	DISASSEMBLE AND REASSEMBLE PRIMARY MANWAY SG-1B
92032858-01	A	DISASSEMBLE AND REASSEMBLE PRIMARY MANWAY SG-1A
92032858-05	A	REPLACED SNUBBER 1-R-NI-1314, REPLACED SNUBBER, LOAD STUD LOCK NUT, EXT PIECE SCREWS, 1-R-NI-1314
92032869-01	A	DISASSEMBLED SG ID MANWAY COVER (REMOVE AND REPLACE)
92033968-01	A	VALVE 1NC001 (REPLACED)
92046405-01	B	VALVE 1CF053 (REPLACED BONNET)
92050396-02	B	REPLACED LOAD STUD, 1-R-NI-107
92052097-01	B	REPLACE VALVE 1NV307, 3"-CUT-OUT WELDS, 1NV39-10,11 (VN 3773)
92052373-01	B	REPLACED SNUBBER 1-R-ND-0391
92052913-01		REPLACED STUDS NUTS, 1ND1-MJ1
92054065-01	B	REPLACED LOAD BOLT, 1-R-SM-1011
92054349-01	B	REPLACED PIN IN REAR BRACKET, 1-R-CF-1567
92054350-01	B	REPLACE SNUBBER 1-R-SM-1565, REPLACED PIN IN REAR BRACKET, 1-R-CF-1567
92055498-01	B	VALVE 1NV223 (REPLACED)
92055879-01	A	DISASSEMBLE AND REASSEMBLE VALVE 1NC27
92060161-01	A	SG ID ACCESS HOLE-MACHINED
92060945-01	A	REPLACED VALVE INI119 NEW SERIAL# TD89439 ITEM# CSR-128
92062213-01	A	REPLACED SNUBBER, 1-R-NC-1655

REPAIR/REPLACEMENT LOG
ASME SECTION XI 1981
U1EOC6

92065299-01	A	VALVE 1NC32B (NO PRESSURE BOUNDARY PARTS REPLACED)
92072880-01	A	DISASSEMBLED REACTOR COOLANT PUMP 1C (NO PRESSURE BOUNDARY PARTS REPLACED)

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8.0 ABSTRACT OF CONDITIONS NOTED AND CORRECTIVE ACTIONS TAKEN

8.1 Refuel-01

8.1.1 RPV Internals

During inspection of the RPV internals/internal components several conditions were reported to Detroit Edison for review/disposition. The reported indications are listed as follows:

- o Tack weld on feedwater sparger bracket at 180° for attachment nut/pin was not visible.
- o Unusual surface conditions (arc strikes and pitting) were noted on Loop A Core Spray piping at approx. 140°. Additional light scratches were noted on both Loop A and Loop B Core Spray internal piping.
- o Small arc strikes were noted on the Core Spray internal piping/sparger brackets at 15° and 150°.
- o A small arc strike was noted on the upper Core Spray sparger (shroud area) at 145°.

The above conditions were evaluated using prudent engineering practices and were determined not to be nonconforming to the original design requirements or detrimental to continued service.

No corrective action was taken to repair these indications. These areas will be monitored during future inspections of the RPV internals as required by ASME Section XI Table IWB-2500-1 (B13.10).

8.1.2 Component Supports

Hanger T48-2097-G21 was found to have insufficient clearances. Deviation Event Report (DER) 89-1315 was initiated for evaluation. It was determined that this was not reportable. The hanger was reworked to provide acceptable clearances as specified on the hanger sketch. Additional adjacent supports were visually inspected with no discrepancies identified.

8.2 Refuel-02

8.2.1 RPV Internals

During inspection of the RPV internals/internal components an additional indication to the ones previously identified during RF-01 was reported to Detroit Edison for review/disposition. The reported indication is listed below:

- o Apparent arc strike was noted on core spray internal piping at 310° this was not recorded in the previous inspection.

This condition and those previously identified during RF-01 were evaluated using prudent engineering practices and were determined not to be nonconforming to the original design requirements or detrimental to continued service.

No corrective action was taken to repair these indications. These areas will be monitored during future inspections of the RPV Internals as required by ASME Section XI Table IWB-2500-1 (B13.10).

8.2.2 Piping Welds

During the inspection of piping welds, 2 welds having rejectable indications were reported to Detroit Edison for review/disposition. The welds are identified below:

- o Weld SW-E11-3151-1WH had rejectable surface indications identified during the magnetic particle examination. DER 91-0262 was initiated for evaluation.
- o Weld SW-RD-2-B3-W5LU-B had rejectable surface indications identified during the liquid penetrant examination. DER 91-0234 was initiated for evaluation.

Both welds were subsequently blend ground to remove the indications and reexamined by both surface and Volumetric techniques with acceptable results. The initial indications on both welds were most likely left over from construction. No additional welds were inspected as a result of these minor indications.

Note:

8.3 Refuel-03

8.3.1 RPV Internals

During inspection of the RPV Internals/Internal components two cracks were reported to Detroit Edison for review/disposition. The reported conditions are listed as follows:

- o Crack Number 1 was located in hood to end plate weld HE-B-1. The crack was approximately 50" long, with a maximum gap of 1/2 inch. The crack ran through the throat of the weld and was caused by high cycle fatigue. This crack is not uncommon to the industry, having occurred at other plants.
- o Crack Number 2 is located in the end plate of dryer bank "A" just above the weld to the end plate of the drain trough. The crack is in the weld heat affected zone (HAZ) between Tie Rods TR-A-7 and TR-A-8. The crack is caused by Intergranular Stress Corrosion Cracking (IGSCC).

Crack Number 1 was repaired by grinding out the existing failed weld and preparing the base metal edges for the new weld, clamping the crack closed, rewelding the hood to end plate joint, and welding a new reinforcing plate over the replaced/existing weld. With the exception of the original failed weld repair, this repair process was repeated at three (3) similar locations where the potential future weld failure was high. This was performed as a preventive measure to preclude future joint failure, higher personnel exposure, and higher future repair costs.

An evaluation was performed on Crack Number 2, and it was determined that this crack did not require repair as there is a low probability that this crack will propagate into weld or base metal outside the HAZ. The crack will tend to grow at a slow rate as the stresses at this crack location during dryer operation are low. Crack Number 2 will continue to be monitored during future outages.

The indications previously identified during inspections performed in RF-01 and RF-02 were again reinspected with no change in conditions noted. These areas in addition to the cracks identified and repairs performed during RF-03 will be monitored during future inspections of the RPV Internals as required by ASME Section XI Table IWB-2500-1 (B13.10).

8.3.2 Component Supports

Several hangers were found with discrepancies between the installed condition and their configuration documents. Deviation Event Report (DER) 92-0573 was initiated for evaluation. It was determined that their nature was such that it did not effect the components operability and was not reportable. No additional supports were inspected as a result of these observations.

FERMI

**NDE ENGINEERING ANALYSIS & APPLICATIONS
INEL/EG&G IDAHO
IDAHO FALLS, IDAHO 83415-2209**

Telecopier Telephone: 208/526-0690
Verification Telephone: 208/526-6122

DATE: February 24, 1993

TO: Winston Liu, NRC RES

FAX: 301 492 3696

FROM: Jeff Cook, Sr.

TELEPHONE: 208 526 6414 office
208 522 7339 home

ASME Section XI Code Violation, Task 24

Per our telecon the following is provided for you to discuss with NRR. The licensee is using the 80-WB1 Section XI. The violation was noted during review of Paragraph 8.2.2 of the Summary Report. Both IWB-2430 and IWC-2430 require additional examinations when rejectable indications are revealed. Note that the licensee characterized the indications as "minor."

Number of Pages (including cover page): 3

Georgia Power

HL-2071
003002

February 26, 1992

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNIT 1
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
INSERVICE INSPECTION ASSOCIATED
WITH THE FALL 1991 MAINTENANCE/REFUELING OUTAGE

Gentlemen:

Pursuant to the requirements of Article IWA-6000 of the ASME Code, Section XI, enclosed for your review is the Owner's Data Report for Inservice Inspection (Form NIS-1) for the inservice inspection activities conducted at Plant Hatch Unit 1 during the Fall 1991 maintenance/refueling outage. The Owner's Data Report is included as Enclosure 1.

Georgia Power Company (GPC) has performed several inspections not specifically required by ASME Code Section XI, on selected Class 1, 2, and 3 components. The results of the Code and augmented inspections performed at Unit 1 are summarized below and reported in the Owner's Data Report.

- o Ultrasonic examinations of austenitic stainless steel piping welds in the primary coolant pressure boundary per Generic Letter 88-01 and NUREG-0313, Revision 2. Results are discussed in the Enclosure under "Class 1 Examinations." A detailed report on flaw evaluations and weld overlay designs was submitted to the NRC via letter dated January 21, 1992.
- o Ultrasonic examinations of eight (8) welds in the nonsafety-related portion of the reactor water cleanup (RWCU) piping were performed per commitments to the NRC related to Generic Letter 88-01. No reportable indications were detected.
- o Ultrasonic thickness measurements were performed on selected components in the extraction steam and condensate/feedwater system piping per GPC's ongoing erosion/corrosion program. A portion of these components were selected due to their similarity in design and operating conditions to components involved with the "Surry pipe break incident".

1/20/92

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11

U.S. Nuclear Regulatory Commission
February 26, 1992
Page Two

- o Eddy current testing was performed on several feedwater heaters, Diesel Generator 1A & 1C coolers, and motor generator "A" air cooler.
- o The core spray sparger and associated piping was visually examined per Bulletin 80-13. No reportable indications were detected.
- o One shroud support access hole cover weld was ultrasonically examined per GE SIL-462. No reportable indications were detected.

If you have any questions in this regard, please contact this office at any time.

Sincerely,

J. T. Beckham, Jr.

SRM/ld
003002

Enclosure: Owner's Data report for Inservice Inspection - Edwin I. Hatch
Nuclear Plant Unit 1 - September 1991 - November 1991

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission
Mr. S. D. Ebnetter, Regional Administrator
Mr. L. D. Wert, Senior Resident - Hatch

References:

1. Letter, J. T. Beckham to NRC, IGSCC Flaw Evaluations and Weld Overlay Designs Unit 1, Fall 1991 Maintenance and Refueling Outage, HL-2000, Dated January 21, 1992.

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NOTE: Portions of this report are compiled from Southern Nuclear Operating Company issued report; "Nondestructive Examination Of Selected Class 1, 2, and 3 Components", for the Fall 1991 Refueling Outage at E.I. Hatch Nuclear Plant, Unit 1. This report is available for review in the Records Management Department at the plant site.

TOTAL NUMBER OF SHEETS 104

FORM NIS-1

FORM NIS-1 OWNERS' DATA REPORT FOR INSERVICE INSPECTIONS
As Required By the Provisions Of The ASME Codes Rules

1. Owner: Georgia Power Company, 333 Piedmont Ave., NE, P.O. Box 4545, Atlanta, Georgia 30302
2. Plant: Edwin I. Hatch Nuclear Plant, Route 1, Box 278, Baxley, GA 31513
3. Plant Unit 1 4. Owner Certificate of Authorization (if req.) N/A
5. Commercial Service Date 12/31/75 6. National Board No. for Unit N/A

7. Components Inspected:

<u>Component or Appurtenance or System</u>	<u>Manufacturer or Installer</u>	<u>Manufacturer or Installer Serial No.</u>	<u>State or Province Number</u>	<u>National Board No.</u>
<u>Rx. Pressure Vessel</u>	<u>Combustion Eng.</u>	<u>67105</u>	<u>N/A</u>	<u>20769</u>
<u>Rx. Pressure Vessel</u>	<u>Combustion Eng.</u>	<u>67205</u>	<u>N/A</u>	<u>20769</u>
<u>1B21 Main Steam</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1B21 Feedwater</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1B31 Rx. Recirc</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1C11 CRD</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1C41 SBLC</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1E11 RHR</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1E21 Core Spray</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1E41 HPCI</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1E51 RCIC</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1G31 RWCU</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1G41 FPC & CU</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1N11 M S Auxiliary</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>
<u>1P41 Plt Serv Water</u>	<u>Pullman Power Prod.</u>	<u>*</u>	<u>N/A</u>	<u>N/A</u>

Note: Supplemental sheets in form of lists, sketches, or drawings may be used provided (1) size is 8 1/2 in. x 11 in., (2) information in items 1 through 6 on this data report is included on each sheet, and (3) each sheet is numbered and the number of sheets are recorded at the top of this form. **

* - Spool piece of fitting numbers too numerous to list for each specific system. Material certifications for all piping, fittings, etc., are available for review in the Records Management Department at the plant site.

** Exception taken to note 2.

ABSTRACT

An Inservice Inspection of selected Class 1, 2 and 3 components at Georgia Power Company's Edwin I. Hatch Nuclear Plant Unit 1 was performed during the Fall 1991 Maintenance/Refueling Outage. The components were examined in accordance with the applicable ISI Outage Plan, including any changes made during the outage as approved by GPC.

Edwin I. Hatch Unit 1 is currently in the second period of the second 10-Year Inspection Interval. The required examinations are presently on schedule as specified in the Second Ten-Year Inspection Plan.

The nondestructive examinations were performed using VT, PT, MT and UT examination techniques. SNC personnel and their contractors; LMT, GE, Seimens, and Ebasco performed NDE of the selected welds and components. In addition, GE assisted SNC personnel with VT examination of selected RPV internal components. SNC, GE, or GPC NDE procedures were utilized for all ASME Section XI Examinations. LMT, Ebasco, and Seimens personnel were qualified to the applicable SNC procedures. EPRI qualified inspectors were utilized for all examinations involving IGSCC susceptible materials. SNC and GE procedures were used for mechanized ultrasonic examination and exams were performed by SNC, Seimens, Ebasco, and GE inspectors.

SNC and C&L personnel performed eddy current examinations of non ASME Section XI components per a request from GPC. C&L procedures were utilized for the performance of the eddy current examinations.

In addition to NDE testing of Class 1 and 2 welds and components, pressure testing, visual examination of Class 1 component internal surfaces and visual examination of pipe supports and hangers were also performed. Third party review (e.g. an ANII) was utilized for all examinations of ASME Section XI components.

Selected components were examined in accordance with GPC commitments to the following documents:

- Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1980 Edition with Addenda through Winter 1981.
- United States Nuclear Regulatory Commission, Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping" which invokes much of NUREG 0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping".
- United States Nuclear Regulatory Commission, Generic Letter 81-11, which modifies and invokes NUREG 0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking".
- United States Nuclear Regulatory Commission, I&E Bulletin 80-13 Visual Examination of Core Spray Spargers.
- SNC "Inservice Inspection Outage Plan, Edwin I. Hatch Nuclear Plant, Unit 1 1991 Fall Refueling Outage, Revision 1."
- SNC "Second Ten-Year Examination Plan, Edwin I. Hatch Nuclear Plant Unit 1."
- United States Nuclear Regulatory Commission NUREG 0803, "Generic Safety Evaluation Report Regarding Integrity of BWR SCRAM System Piping."

Representative samples of the following systems, comprised of selected Class 1, 2, and 3 components, were examined using various NDE techniques, in accordance with the above documents:

Class 1

Reactor Pressure Vessel (1B11)
Main Steam System (1B21)
Feedwater System (1B21)
Reactor Recirculation System (1B31)
Control Rod Drive System (1C11)
Standby Liquid Control System (1C41)
Residual Heat Removal System (1E11)
Core Spray System (1E21)
High Pressure Coolant Injection System (1E41)
Reactor Core Isolation Cooling System (1E51)
Reactor Water Cleanup System (1G31)
Valve Internals
Valve Bolting

Class 2

Standby Liquid Control System (1C41)
Residual Heat Removal System (1E11)
Core Spray System (1E21)
High Pressure Coolant Injection System (1E41)
Reactor Water Cleanup System (1G31)
Reactor Core Isolation Cooling System (1E51)
Main Steam Auxiliary System (1N11)

Class 3

RHR Service Water System (1E11)
Fuel Pool Cooling System (1G41)
Plant Service Water System (1P41)

Other - Augmented (Non ASME Section XI)

Eddy-current examinations were performed on the following components:
Unit 1 Feedwater Heaters 5A, 5B, 7A, 7B, 8A, 8B, 10A, 10B, and Emergency Diesel Generator 1A and 1C Jacket Water Coolers, Lube Oil Coolers, Air Coolers and the Motor Generator "A" Air Cooler.

UT thickness measurements were performed on selected components in the Extraction Steam and Condensate Feedwater Piping Systems. A portion of these components were selected due to their similarity in design and operating conditions to components involved with the "Surry pipe break incident".

Eight (8) Non-Safety RWCU System welds were examined using ultrasonic examinations techniques (UT) due to commitments made by GPC. These augmented exams are not required by the ASME Section XI Code but were performed due to commitments to GL 88-01.

CLASS 1 EXAMINATIONS

NUREG 0313

GPC is committed to the performance of surface and volumetric examinations on IGSCC susceptible welds in accordance with NUREG 0313. This commitment is formalized in GPC response to NRC Generic Letter 88-01. The below listed summary gives the total number of exams performed by outage end. A detailed report on flaw evaluation and overlay design was submitted to the NRC by letter HL-2000 dated January 21, 1992.

Category A

Thirty-nine (39) Category A welds were examined. Twenty-five (25) of these welds were examined using UT and PT techniques and the remaining fourteen (14) were examined using UT techniques only. No rejectable indications were found.

Category C

The original examination scope included fourteen (14) Category C welds. A scope expansion was required due to one (1) weld being found unacceptable for continued operation and was overlay repaired, resulting in a total of twenty-eight (28) Category C welds being examined. Fourteen (14) of these welds were examined using UT and PT techniques and the remaining fourteen (14) welds were examined using UT techniques only.

The Category C weld which was overlay repaired was 1B31-1RC-28B-2.

~~Although weld 1B31-1RC-28B-2 did not contain any identified flaws, a weld overlay was applied to improve the inspectability of the location.~~

The classification of these welds is now Category E per NUREG-0313 and will be examined as such in future outages.

Category D

Twelve (12) Category D welds were examined. No rejectable indications were detected. Five (5) of these welds were examined using UT and PT techniques and the remaining seven (7) were examined using UT techniques only.

Six (6) RINTSA welds were examined by UT and no rejectable indications were detected.

Category E

The original examination scope included twenty-five (25) of the total forty-six (46) Category E (overlayed) welds. UT examinations revealed new IGSCC indications and some original cracks with new growth which warranted examination scope expansion. By outage end, all forty-six (46) Category E welds had been examined by UT NDE techniques. All results for new indications or crack growth were evaluated by Structural Integrity Associates, Inc. and found to be acceptable for continued service.

Note that six (6) new overlays were added during this outage which results in a total of fifty-two (52) Category E overlays. Preservice examinations were performed on each of these six (6) new overlays. No rejectable indications were found. See sections on Category C and Category F examinations for the additional overlay weld numbers.

Category F

All Category F welds require examination each refueling outage. All four (4) of these welds were examined during the outage and a change in indications or apparent crack growth was detected in all four (4) of these welds which were unacceptable for continued service. All four (4) welds were overlay repaired and will be examined as Category E welds in future outages. Listed below are the Category F welds which were overlay repaired:

1B31-1RC-12BR-A-4
1B31-1RC-12BR-E-4
1B31-1RC-12AR-G-4
1E11-1RHR-20B-D-4

A preservice baseline examination was performed on all welds which were overlay repaired during the outage.

Other Class 1 Examinations

Two-hundred-eighteen (218) ASME Section XI component examinations were performed utilizing UT, MT, PT and VT as applicable. These examinations included; RPV welds, piping welds, valve internals, and valve bolting materials.

Per I&E Bulletin 80-13, the core spray sparger and associated piping were VT examined. No reportable indications were detected.

Per ASME Section XI, selected RPV internals were examined. These examinations included portions of the vessel interior, interior attachments beyond the belt line region and the RPV internal components. See the In-vessel Inspection section of this report for more detailed information.

Four (4) Class 1 valves were disassembled for maintenance/inspection during the outage. The internals of these valves were VT inspected by GPC Q.C. personnel. None of the valves exhibited any unacceptable conditions relevant to the visual examination.

Twenty (20) CRD's were replaced during the outage which facilitated visual examination (VT-1) of the bolts, studs, and nuts.

Per SIL 462, one Shroud Support Access Hole Coverplate was examined.

Class II Examinations

Forty-seven (47) welds were examined using surface and/or volumetric NDE techniques as applicable. Two (2) of these welds were examined per NUREG 0619 (UT only), eleven (11) examinations were for the GPC augmented examination commitments, and the remaining thirty-four (34) examinations were per ASME Section XI requirements.

Unacceptable linear indications were found in three (3) welds and were removed by controlled grinding.

Pressure Testing

One (1) Class 1 hydrostatic test, two (2) Class 2 hydrostatic tests, three (3) Class 2 functional tests, two (2) Class 3 hydrostatic tests, three (3) Class 3 inservice tests, and the Class 1 System Leakage Test were all performed satisfactorily. See Pressure Test Section of Report for specific test identifications and details.

Augmented Examinations

Eight (8) welds in the non-safety 3 portion of the RWCU System were examined by UT per a GPC commitment to the NRC for NUREG 0313 Rev. 2 augmented requirements. No reportable indications were detected.

Component Support Examinations (Class 1, 2 and 3)

One hundred and forty-nine (149) component supports were VT examined per the requirements of ASME Section XI during the outage. Seventeen (17) produced unacceptable results. After maintenance and/or engineering evaluation, all of the unacceptable component supports were determined to be acceptable. Where maintenance was required, the component supports were re-examined to confirm acceptability.

Repairs and Replacements (Class 1 and 2)

Numerous repair/replacement activities were performed prior to and during the outage. Major repair/replacement activities included weld overlay of six (6) Reactor Recirc. welds. An itemized list of the repair/replacement activities is included in the Repair/Replacement Section of this report.

Reportable Indications

Following is an itemized list of all welds and components which were reported with indications or were considered unacceptable. All of these items were either repaired and/or evaluated and then determined to be acceptable.

Summary of Indications

<u>Identification</u>	<u>Indication</u>	<u>Corrective Action</u>
Steam Dryer Upper and Lower Guides	Rub marks, displaced metal	Acceptable as is per GPC and GE engineering
Upper Support Bracket 1E11-2HX-A-USC-2	.3" linear indication	Indication removed, area re-welded MWO 1-91-5805
Upper Support Bracket 1E11-2HX-A-USC-3	.6" and .4" linear indications	Indication removed, area re-welded MWO 1-91-6395
Weld 1B31-1RC-12BR-D-3	Spot indication	Acceptable as is per GPC Engineering
Weld overlay 1B31-1RC-28A-7	6 interbead lack of fusion indications	Acceptable as is per SIAI
Weld overlay 1B31-1RC-28B-15	.3" linear indication	Acceptable as is per SIAI
Weld overlay 1E11-1RHR-24B-R-12	7 interbead lack of fusion indications	Acceptable as is per SIAI
Weld 1B31-1RC-28B-2	.5", .42", and .25" linear indications	Weld overlay MWO 1-91-4832
Weld 1E11-2RHR-20A-D-4	3.5" linear MT indication	Indication removed, area re-welded MWO 1-91-6265
Weld 1E11-1RHR-24B-R-7	2.5" linear MT indication	Indication removed, area re-welded MWO 1-91-6264
Weld 1B31-1RC-12BR-A-4	UT indication	Weld overlay MWO 1-91-4832
Weld 1B31-1RC-12BR-E-4	UT indication	Weld overlay MWO 1-91-4832
Weld 1B31-1RC-12AR-G-4	UT indication	Weld overlay MWO 1-91-4832
Weld 1E11-1RHR-20B-D-4	UT indication	Weld overlay MWO 1-91-4832
Support 1B21-FDH-5	Broken fillet welds	Acceptable as is per Bechtel letter
Support 1B21-FDH-10	Bent paddle, slipped bearing	Replaced paddle, reset bearing MWO 1-91-6468

<u>Identification</u>	<u>Indication</u>	<u>Corrective Action</u>
Support 1C11-SK1-H3	Bent rod	Acceptable as is per Bechtel letter
Support 1C11-SK1-H5	No load on support	Readjusted rod MWO 1-91-6504
Support 1C11-SK1-H6	Light load on support	Readjusted rod *MWO 1-91-6570
Support 1C11-SK1-H14	Bent rod	Acceptable as is per Bechtel letter
Support 1C11-SK2-H24	No load on clevis	Readjusted rod *MWO 1-91-6570
Support 1C11-SK2-H26	No load on clevis, loose nut	Readjusted rod, tightened loose nut *MWO 1-91-6570
Support 1E11-RHRH-407	Slipped bearing	Reset bearing MWO 1-91-6129
Support 1E11-RHRH-407A	Slipped bearing	Reset bearing MWO 1-91-6129
Support 1E11-SM-1	Restricted movement of paddle	Acceptable as is per Bechtel letter
Support 1P41-SDGH-4	Improper spring can setting	Reset spring can MWO 1-91-5455
Support 1P41-SDGH-7	Improper spring can setting	Reset spring can MWO 1-91-5455
Support 1P41-SWH-19	Bent rod	Repaired rod MWO 1-91-5456
Support 1P41-ISH-33	Bent paddle, slipped bearing	Replaced paddle, bearing acceptable as is per Bechtel letter MWO 1-91-5454
Support 1E11-RHRH-60	Slipped bearing	Reset bearing MWO 1-91-5457
Support 1E41-HPSEH-78	Improper spring can setting	Reset spring can *MWO 1-91-5458
Weld 1B31-1RC-12AR-H-3	3.8" and 1.3" linear indications	Acceptable as is per SIAI

<u>Identification</u>	<u>Indication</u>	<u>Corrective Action</u>
Weld 1B31-1RC-12AR-J-3	1.3" linear indication	Acceptable as is per SIAI
Weld 1E11-1RHR-24B-R-13	.4", .15", .25", .6", .6" and .3" linear indications	Acceptable as is per SIAI

* These MWOs did not require initial/final ANII review.

E.I. HATCH UNIT 1 FALL 1991 REFUELING OUTAGE
CLASS 1 COMPONENTS

ASME SECTION XI	EXAM FIGURE NO.	EXAMINATION/AREA	EXAMINATION PROCEDURE	CAL BLOCK	EXAM/CAL SHEET NO.	RESULTS	REMARKS
<u>RPV EXAMINATIONS</u>							
R1.10 R H ASME	A-1A/02	C-6 (120-270) SUPPORT SKIRT-N2D (120 DEGREE) TO N2H (270 DEGREE) C.W.	MT-H-500/05 UT-H-410/04	61-H	S91H1M044 S91H1C106 S91H1U157	NRI N/A UT CAL NRI	N/A
R15.10 R P ASME	-	CLASS 1 (PT) PRESSURE RETAINING BOUNDARY LEAKAGE TEST	VT-H-720/03	N/A			SEE PRESSURE TEST SECTION OF THIS REPORT.
H7.80 H G-2 ASME	---	FLANGE BOLTING CRD HOUSING	45QC-INF- 012-0S	N/A			EXAM PERFORMED BY GPC QC DEPT.
NUREG-0113D	-	N2A (RINTSA) RINTSA WELD	UT-H-415/05	125-H	S91H1C169 S91H1U231	N/A UT CAL NRI	N/A
R1.100 R-D ASME	A-1/04	N2B (IR) B LOOP RECIRCULATION INLET NOZZLE AT 60 DEGREES	UT-H-480/04	61-H	S91H1C104 S91H1U155	N/A UT CAL NRI	EXAM LIMITATION DUE TO WELDED WELD 93% COVERAGE <i>Cold Case</i>
R13.90 R-D ASME	A-1/04	N2B (N-SH) B LOOP RECIRCULATION INLET NOZZ TO SHELL	UT-H-410/04	61-H	S91H1C115 S91H1U167 S91H1C118 S91H1U171 S91H1C121 S91H1U175	N/A UT CAL NRI N/A UT CAL NRI N/A UT CAL NRI	ONE-SIDED EXAM DUE TO NOZZLE CONFIGURATION

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

Plant H

January 14, 1993

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Braidwood Station Unit 1
Inservice Inspection Report for
Interval 1, Period 2, Refuel Outage 1
NRC Docket Nos. 50-456

Reference:(a) December 15, 1988, S.C. Hunsader to letter
to T. E. Murley

Dear Dr. Murley:

Reference (a) provided the complete Braidwood Station Units 1 and 2 Inservice Inspection (ISI) program for NRC review that is currently being implemented at the station. In accordance with the requirements of this program and Article IWA-6000 of Section XI of the ASME Boiler and Pressure Vessel Code (1983 Edition through and including the Summer, 1983 Addenda) Commonwealth Edison is submitting the report of the ISI performed at Unit 1 from May 19, 1991 through November 3, 1992. This report is required to be submitted within 90 days of the completion of a refueling outage. The Unit 1 refueling outage was completed on November 3, 1992.

Please address any questions regarding this submittal to this office.

Very truly yours,

T.W. Simpkin
Nuclear Licensing Administrator

190024

cc: J. Hickman - NRR w/o attachment
B. Clayton - RIII w/o attachment
Braidwood Resident Inspector w/o attachment

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AD47

COMMONWEALTH EDISON
BRAIDWOOD NUCLEAR GENERATING STATION
UNIT 1
INSERVICE INSPECTION SUMMARY REPORT
INTERVAL 1
PERIOD 2
OUTAGE 1

STATION ADDRESS:

BRAIDWOOD NUCLEAR GENERATING STATION

R.R. #1 BOX 84

BRACEVILLE, ILLINOIS 60407

COMMERCIAL SERVICE DATE: JULY 29, 1988

OWNER'S ADDRESS:

COMMONWEALTH EDISON COMPANY

P.O. BOX 767

CHICAGO, ILLINOIS 60690

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INTRODUCTION

Braidwood Station performed Inservice Inspections on ASME Class 1 and 2 components and piping for Unit 1 from May 19, 1991 to November 03, 1992. The majority of the inspections were performed during Braidwood Station's Unit 1 third refueling outage (A1R03). Examinations were performed in compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 1983 Edition through the Summer 1983 Addenda of Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", as required by the Code of Federal Regulations (10CFR50.55a). This summary report addresses the inspection results on Class 1 and 2 components and piping.

Braidwood's Nondestructive Examination (NDE) ISI Program Plan for Class 1 and 2 components and piping was developed in accordance with ASME Section XI, Division 1, Subsections IWA, IWB, and IWC. In addition to the ASME Section XI examination requirements, certain Nuclear Regulatory Commission (NRC) augmented inspections were included. Braidwood's Unit 1 augmented ISI requirements included:

- Examination of High Energy Lines in compliance with NUREG 0800, Section 6.6, "Standard Review Plan".
- Regulatory Guide 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes".
- NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors".
- NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems".

Braidwood developed and submitted specific relief requests, for the Commission's review, where Code and/or augmented requirements were deemed impractical.

Identification of Examination Requirements

The ISI Program contains NDE tables, presented in a tabular format, consistent with the ASME Section XI Subsections IWB and IWC tables. The NDE program tables include the corresponding code category, item number, and component/weld selection in compliance with ASME Section XI Class 1 and 2 examination requirements.

Exempted Components

The ISI NDE Program Plan identifies ASME Class 1 and 2 components exempt from examination as specified in ASME Section XI, Subsections IWB and IWC, Paragraph 1220, "Components Exempt from Examination" along with technical justifications for these exemptions.

ISI Program Implementation

Braidwood Station performed visual examinations (VT-1, VT-2, and VT-3/4) on ASME Class 1 and 2 components and their supports identified in the ISI program due to repairs or replacements under ASME Section XI requirements. The examinations were performed by personnel qualified to Commonwealth Edison's "Special Process Procedure (SPP 2-1-0)" developed in compliance with the American Society of Nondestructive Testing (SNT-TC-1A), 1984 Edition, and American National Standards Institute (ANSI), 1973/1978 Edition, N45.2.6.

All NDE Examinations, including evaluations of the results, were performed by personnel qualified to the American Society of Nondestructive Testing (SNT-TC-1A) and in compliance with ASME Sections V and XI, 1983 Edition through Summer 1983 Addenda.

NDE EXAMINATIONS

Braidwood performed all the ISI nondestructive examinations, including the evaluations of flaw indications, in accordance with ASME Section XI, subarticle IWA-2200, "Examination Methods".

Inservice Inspections were performed from May 19, 1991 to November 3, 1992. The Inspector was Mr. Librado Malabanan from Hartford Steam Boiler Inspection and Insurance Company of Hartford, Connecticut, Chicago Branch, 2443 Warrenville Road, Suite 500 Lisle, Illinois 60532.

The following is a summary of the examinations performed during the Braidwood Unit 1 third refueling outage. Refer to the summary tables for additional information on specific welds, components, supports and pressure test examinations and their respective results.

PIPING SYSTEMS

CHEMICAL AND VOLUME CONTROL (CV)

A total of 2 augmented pressure retaining welds were examined (ref. Note 10 of the ISI Program Plan). One valve received a VT-1 exam (B7.70)

FEEDWATER (FW)

A total of 23 pressure retaining welds were examined. Of these 12 received a volumetric and surface exam (C5.21) and 11 received a volumetric exam only (ref. Note 5 of the ISI Program Plan).

MAINSTEAM (MS)

A total of 20 weldments were examined. Of these 4 received a surface exam (2-C3.20, 2-C5.22), 4 received both a volumetric and surface exam (C5.21) and 12 received a volumetric exam only (ref. Note 5 of the ISI Program Plan).

REACTOR COOLANT (RC)

A total of 59 components were examined. Of these 18 welds were examined by surface & volumetric (15-B9.11, 2-B5.70, 1-B9.31), 35 welds received a surface exam (28-B9.40, 1-B9.21, 6-B9.32), and 6 mechanical bolting connections received a VT-1 (3-B7.50, 3-B7.70).

RESIDUAL HEAT REMOVAL (RH)

A total of 12 pressure retaining welds received a surface exam (3-C5.12, 9-C5.11) and 8 received a volumetric examination (ref. Note 10 of the ISI Program Plan).

SAFETY INJECTION (SI)

A total of 6 pressure retaining welds were examined. Four welds received a volumetric and surface exam (B9.11), and 2 welds received a surface exam (B9.40).

COMPONENTS

REACTOR VESSEL (1RC01R)

A VT-3 was performed on the accessible surfaces of the vessel (B13.10).

REACTOR COOLANT PUMP (1RC01PD)

1 reactor coolant pump flywheel was examined per Reg Guide 1.14 (ref. Note 6 of the ISI Program Plan).

RESIDUAL HEAT REMOVAL HEAT EXCHANGER (1RH02AA,1RH02AB)

Four nozzle-to-shell (C-B C2.21) pressure retaining welds received a volumetric and surface exam.

STEAM GENERATOR (1RC01BA)

One volumetric examination was performed (B2.40).

STEAM GENERATORS (1RC01BA, 1RC01BB, 1RC01BC, 1RC01BD)

Eddy Current examination was performed on all 4 steam generators. Fifty percent of each generator's tubes received 100% full length examination. The remaining tubes were examined from the hot leg tube end thru the top support plate on the cold leg side. Based on Eddy Current results, 174 tubes were plugged.

Additional information concerning the steam generator eddy current inspection can be obtained from the report submitted to the Nuclear Regulatory Commission as required by Braidwood Station Technical Specifications 4.4.5.5.

SNUBBERS/COMPONENT SUPPORTS

<u>SYSTEM</u>	<u>SNUBBER VT-3</u>	<u>COMP. SUPPORT VT-3</u>
#NSSF	8	.
AF	2	
CS	4	17
CV	56	24
FW	4	12
MS	12	
RC	94	9
RH	39	17
RY	33	4
SD	15	
SI	65	48
	<hr/>	<hr/>
	332	131

#STEAM GENERATOR SNUBBERS

V

**SYSTEM PRESSURE TESTS
EXAMINATION SUMMARY
TABULATION**

VI

C-111

SYSTEM PRESSURE TESTS

A total of 1 Class 1 System Pressure Test and 68 Class 2 Subsystem Pressure Tests were performed.

The Class 1 System Boundary includes the Class 1 portions of Chemical Volume Control System (CV), Reactor Coolant (RC), Residual Heat Removal System (RH), Pressurizer System (RY), and the Safety Injection System (SI).

The Class 2 system boundaries are divided into convenient test boundaries and are described briefly below:

<u>SUBSYSTEM</u>	<u>CLASS</u>	<u>DESCRIPTION</u>
CC-1	2	Component Cooling IRC/ORC
CC-2	2	Component Cooling IRC/ORC
CS-1	2	1A Train Containment Spray System
CS-2	2	1B Train Containment Spray System
CS-3	2	1A Train Spray Additive System
CS-4	2	1B Train Spray Additive System
CV-1	2	Volume Control Tank
CV-2	2	Positive Displacement Pump
CV-3	2	1A Centrifugal Charging Pump
CV-4	2	1B Centrifugal Charging Pump
CV-5	2	Control Volume Charging Header
CV-6	2	1A Seal Water Injection Filter
CV-7	2	1B Seal Water Injection Filter
CV-8	2	Seal Water Injection
CV-9	2	Seal Water Return
CV-10	2	Chemical Volume Charging to RC Loops
CV-11	2	Charging Pumps Regenerative Heat Exchanger
CV-12	2	1A Regenerative Heat Exchanger
CV-13	2	1B Regenerative Heat Exchanger
CV-14	2	CV Charging Inside Containment
CV-15	2	Letdown Inside Containment
CV-16	2	1B Letdown Regen. Heat Exchanger
CV-17	2	1A Letdown Regen. Heat Exchanger
CV-18	2	Letdown Outside Containment
CV-19	2	1A Letdown Heat Exchanger
CV-20	2	1B Letdown Heat Exchanger
CV-21	2	Letdown Downstream of Letdown Heat Exchanger
CV-22	2	RC Filter up to Volume Control Tank
CV-23	2	1A Excess Letdown Heat Exchanger
CV-24	2	1B Excess Letdown Heat Exchanger
CV-25	2	Letdown Downstream of Excess Letdown Heat Exchanger

FC-1	2	Class 2 Portion of Fuel Pool Cooling
MS-1	2	Mainsteam A Loop
MS-2	2	Mainsteam B Loop
MS-3	2	Mainsteam C Loop
MS-4	2	Mainsteam D Loop
OG-1	2	Hydrogen Recombiner
OG-2	2	Hydrogen Recombiner Suction Piping
OG-3	2	Hydrogen Recombiner Discharge Piping
RC-1	2	Instr./Sampling Lines Loop A & Reactor Vessel
RC-2	2	Instr./Sampling Lines, Loop b
RC-3	2	Instr./Sampling Lines, Loop C
RC-4	2	Instr./Sampling Lines, Loop D
RC-5	2	Reactor Coolant
RH-1	2	1A Train Residual Heat Removal Pump
RH-2	2	1A Train Residual Heat Removal Heat Exchanger
RH-3	2	1B Train Residual Heat Removal Pump
RH-4	2	1B Train Residual Heat Removal Heat Exchanger
RH-5	2	Residual Heat Removal Heat Exchanger
RY-1	2	Pressurizer Sample System
RY-2	2	Pressurizer, Instrumentation lines
RY-3	2	ORC Portion of RY to PRT
RY-4	2	ORC Portion of RY to PRT
SD-1	2	Loop A Steam Gen. Blowdown Lines
SD-2	2	Loop B Steam Gen. Blowdown Lines
SD-3	2	Loop C Steam Gen. Blowdown Lines
SD-4	2	Loop D Steam Gen. Blowdown Lines
SI-1	2	ECCS Suction Piping
SI-2	2	Safety Injection Accumulator
SI-3	2	1A Safety Injection Accumulator
SI-4	2	1B Safety Injection Accumulator
SI-5	2	1C Safety Injection Accumulator
SI-6	2	1D Safety Injection Accumulator
SI-7	2	Accumulator Fill Lines
SI-8	2	Safety Injection To Accumulator Tank
SI-9	2	Safety Injection Test Lines
SX-1	2	ORC/IRC Essential Service Water "B" Train Return Header
SX-2	2	ORC/IRC Essential Service Water "A" Train Return Header

FORM NIS-1 OWNER'S REPORT FOR INSERVICE INSPECTIONS
As required by the Provisions of the ASME Code Rules

1. Owner Commonwealth Edison, One First National Plaza, Chicago, Illinois
(Name and Address of Owner)
2. Plant Braidwood Station, R.R. 1, Box 84, Braceville, IL 60407
(Name and Address of Plant)
3. Plant Unit 1 4. Owner Certificate of Authorization (if required) N/A
5. Commercial Service Date 7-29-1988 6. National Board Number for Unit N/A
7. Components Inspected Class 1 and 2 components as described in Supplemental Sheets (pages 1 thru 18).

Component or Appurtenance	Manufacturer or Installer	Manufacturer or Installer Serial No.	State or Province No.	National Board No.
	See Supplemental Sheets (pages 1 thru 18).			

Note: Supplemental sheets in form of lists, sketches, or drawings may be used, provided (1) size is 8½ in. x 11 in., (2) information in items 1 through 6 on this report is included on each sheet, and (3) each sheet is numbered and the number of sheets is recorded at the top of this form.

WELD/COMPONENT
COMPONENT SUPPORTS
PRESSURE TESTS
EXAMINATION SUMMARY
TABULATION

IX

C-115

CHEMICAL & VOLUME CONTROL

740(172292)/ZD79G

WELD/COMPONENT EXAMINATION NOTES

1. These are augmented volumetric examinations on 7 1/2% of the Circumferential welds within the ECCS systems (ref. Note 10 of the ISI Program Plan).
2. Braidwood Station will utilize Code Case N-426 (ref. Note 3 of the ISI Program Plan).
3. Indications exceeding Section XI acceptance criteria were detected on two (2) nozzle-to-vessel welds. Fracture mechanics analysis was performed and found acceptable. (ref. RC Docket No. STN 50-457).
4. Ultrasonic examinations were performed on high energy lines not postulated for line breaks (ref. Note 5 of the ISI Program Plan).
5. Augmented NDE exams were performed on 1 reactor coolant pump flywheel. (ref. Note 6 of the ISI Program Plan).
6. Only 25% thru wall indications can be detected on cast stainless steel welds (ref. Relief Request NR-2 of the ISI Program Plan).

SYSTEM PRESSURE TESTS

A total of 1 Class 1 System Pressure Test, 68 Class 2 Subsystem Pressure Tests were performed.

The Class 1 System Boundary includes the Class 1 portions of Chemical Volume Control System (CV), Reactor Coolant (RC), Residual Heat Removal System (RH), Pressurizer System (RY), and the Safety Injection System (SI).

The Class 2 system boundaries are divided into convenient test boundaries and are described briefly below:

<u>Subsystem</u>	<u>Class</u>	<u>Description</u>
CC-1	2	Component Cooling IRC/ORC
CC-2	2	Component Cooling/IRC/ORC
CS-1	2	1A Train Containment Spray System
CS-2	2	1B Train Containment Spray System
CS-3	2	1A Train Spray Additive System
CS-4	2	1B Train Spray Additive System
CS-23	2	1A Excess Letdown Heat Exchanger
CS-24	2	1B Excess Letdown Heat Exchanger
CS-25	2	Letdown Downstream of Excess Letdown Heat Exchanger
CV-1	2	Volume Control Tank
CV-2	2	Positive Displacement Pump
CV-3	2	1A Centrifugal Charging Pump
CV-4	2	1B Centrifugal Charging Pump
CV-5	2	Control Volume Charging Header Outside Containment
CV-6	2	1A Seal Water Injection Filter
CV-7	2	1B Seal Water Injection Filter
CV-8	2	Seal Water Injection
CV-9	2	Seal Water Return
CV-10	2	Chemical Volume Charging to Reactor Coolant Loops Inside
CV-11	2	CV Portion of charging Pumps Regenerative Heat Exchanger
CV-12	2	1A CV Regenerative Heat Exchanger
CV-13	2	1B CV Regenerative Heat Exchanger
CV-14	2	CV Charging Inside Containment
CV-15	2	Letdown Inside Containment
CV-16	2	1B Letdown Regen. Heat Exchanger
CV-17	2	1A Letdown Regen. Heat Exchanger
CV-18	2	Letdown Outside Containment
CV-19	2	1A Letdown Heat Exchanger
CV-20	2	1B Letdown Heat Exchanger
CV-21	2	Letdown Downstream of Letdown Heat Exchangers
CV-22	2	RC Filter up to Volume Control Tank
CV-25	2	Letdown Downstream of Excess Letdown Heat Exchanger
FC-1	2	Class 2 Portions of Fuel Pool Cooling

MS-1	2	Main Steam A Loop
MS-2	2	Main Steam B Loop
MS-3	2	Main Steam C Loop
MS-4	2	Main Steam D Loop
OG-1	2	Hydrogen Recombiner
OG-2	2	Hydrogen Recombiner Suction Piping
OG-3	2	Hydrogen Recombiner Discharge Piping
RC-1	2	Instrumentation/Sampling Lines Loop A and Reactor Vessel
RC-2	2	Instrumentation/Sampling Lines, Loop B
RC-3	2	Instrumentation/Sampling Lines Loop C
RC-4	2	Instrumentation/Sampling Lines Loop D
RC-5	2	Reactor Coolant
RH-1	2	1A Train Residual Heat Removal Pump
RH-2	2	1A Train Residual Heat Removal Heat Exchanger
RH-3	2	1B Train Residual Heat Removal Pump
RH-4	2	1B Train Residual Heat Removal Heat Exchanger
RH-5	2	Residual Heat Removal to Safety Injection
RY-1	2	Pressurizer Sample System
RY-2	2	Pressurizer, Instrumentation Lines
RY-3	2	ORC Portion of RY to PRT
RY-4	2	ORC Portion of RY to PRT
SD-1	2	Loop A Steam Gen. Blowdown Lines
SD-2	2	Loop B Steam Gen. Blowdown Lines
SD-3	2	Loop C Steam Gen. Blowdown Lines
SD-4	2	Loop D Steam Gen. Blowdown Lines
SI-1	2	Refueling Water Storage Tank ECCS Suction Piping
SI-2	2	Safety Injection Pumps
SI-3	2	1A Safety Injection Accumulator
SI-4	2	1B Safety Injection Accumulator
SI-5	2	1C Safety Injection Accumulator
SI-6	2	1D Safety Injection Accumulator
SI-7	2	Accumulator Fill Lines
SI-8	2	Safety Injection To Accumulator Tank ORC Portion
SI-9	2	Safety Injection Test Lines
SX-2	2	ORC/IRC Essential Service Water "B" Train Return Header
SX-1	2	ORC/IRC Essential Service Water "A" Train Return Header

SYSTEM PRESSURE TESTS

Class 1

CODE ITEM	EXAMINATION ITEM/AREA	EXAMINATION TYPE				
		LEAK	FUNC.	INSERV	HYDRO	NOTE
B15.10	Class 1 Reactor Vessel Pressure Retaining Boundary	X				1
B15.20	Class 1 Pressurizer Pressure Retaining Boundary	X				1
B15.30	Class 1 Steam Generator Pressure Retaining Boundary:					
	1RC01BA	X				1
	1RC01BB	X				1
	1RC01BC	X				1
	1RC01BD	X				1
B15.50	Class 1 Piping Pressure Retaining Boundary:	X			X	1
B15.60	Class 1 Pumps:	X				1
	1RC01PA					
	1RC01PB					
	1RC01PC					
	1RC01PD					
B15.70	Class 1 Valves Pressure Retaining Boundary	X				1

SYSTEM PRESSURE TESTS

Class 2

CODE ITEM	EXAMINATION ITEM/AREA	EXAMINATION TYPE				
		LEAK	FUNC.	INSERV	HYDRO	NOTE
C2.33	Nozzle-to-Shell (or Head) when inside of vessel is inaccessible: Residual Heat Removal Heat Exchangers: 1RH02AA 1RH-2 1RH02AB 1RH-4			X X		2 2
C7.10	Pressure Vessels Pressure Retaining Components Seal Water Heat Exchangers: 1CV02A 1CV-9 Letdown Heat Exchanger: 1CV04AA 1CV-19 1CV04AB 1CV-20 Seal Water Return Filter: 1CV02F 1CV-9 Residual Heat Removal Heat Exchangers: 1RH02AA 1RH-2 1RH02AB 1RH-4 Volume Control Tank 1CV01T 1CV-1		X	X X X X X		2 2 2 2 2

NIS-2
OWNER'S REPORT FOR
REPAIRS AND REPLACEMENTS

<u>SYSTEM</u>	<u>NIS-2</u>	<u>TOTAL</u>	<u>TOTAL PAGES</u>
Chemical Volume Control (CV)	5		5
Main Steam (MS)	3		3
Reactor Coolant (RC)	4		4
Pressurizer (RY)	3		3
Steam Generator Blowdown (SD)	4		4

**COMMONWEALTH EDISON
BRAIDWOOD NUCLEAR STATION
UNIT 2**

INSERVICE INSPECTION SUMMARY REPORT

INTERVAL 1

PERIOD 1

OUTAGE 2

STATION ADDRESS:

R.R. #1 BOX 84

BRACEVILLE, ILLINOIS 60407

COMMERCIAL SERVICE DATE: OCTOBER 17, 1988

OWNER'S ADDRESS:

COMMONWEALTH EDISON COMPANY

P.O. BOX 767

CHICAGO, ILLINOIS 60690

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INTRODUCTION

Inservice inspections of ASME Class 1 and 2 components and piping were conducted at Braidwood Station Unit 2 from May 28, 1990 to November 26, 1991. The majority of these inspections were performed during Braidwood Stations's Unit 2 second refueling outage (A2R02). The examinations were performed in compliance with the rules and regulations of Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, 1983 Edition through the Summer 1983 Addenda, as per the requirements of Title 10, Part 50.55a of the Code of Federal Regulations (10CFR50.55a). This summary report will address inspection results on Class 1 and 2 components and piping.

The Nondestructive Examination (NDE) ISI Program Plan for Class 1 and 2 components was developed in accordance with the requirements and intent of Subsections IWA, IWB, and IWC of Section XI Division 1, of the ASME Code. In addition to the ASME Section XI requirements of examination, certain augmented inspections were required by the Nuclear Regulatory Commission. The Braidwood Unit 2 augmented ISI requirements included:

- Examination of High Energy lines in conformance with NUREG 0800, "Standard Review Plan", Section 6.6.
- Regulatory Guide 1.83 Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes".
- Regulatory Guide 1.150 Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations".
- Regulatory Guide 1.14 Revision 1, "Reactor Coolant pump Flywheel Integrity".
- NRC Bulletin 88-09 "Thimble Tube Thinning in Westinghouse Reactors".

Where the Code and/or augmented requirement(s) were deemed to be impractical, specific requests for relief were developed and submitted for the Commission's review.

Identification of Examination Requirements

The ISI Program contains the NDE program tables. These tables are presented in a tabular format consistent with Tables IWB and IWC of the ASME Code. The NDE program tables include the corresponding code category, item number, and component/weld selection in conformance with Class 1 and 2 examination requirements. Program notes and relief requests are identified in the remarks column.

Exempted Components

Class 1 and 2 components (or parts of components) which were not included in the NDE program tables and are exempt from examination, as specified in Section XI Paragraph IWB and IWC-1220 "Components Exempt from Examination" are identified in the NDE Program Plan together with the technical justification(s) for exempting the component/system.

Implementation of the ISI Program

Visual examinations (VT-1, VT-2, and VT-3/4) of Class 1 and 2 components and their supports, identified under the ISI Program Plan and/or requiring reexamination(s) due to repairs or replacements under Section XI, were performed by Commonwealth Edison Company, or their designee.

All NDE examinations, including evaluation of results, were performed by personnel qualified to the appropriate levels in the various NDE methods, per the requirements of the American Society for Nondestructive Testing SNT-TC-1A, and in conformance with ASME Sections V and XI, 1983 Edition through Summer 1983 Addenda.

Visual examinations (VT-3 and VT-4) of Class 1 and 2 component supports including evaluation of results, were performed by examiners qualified to ANSI N45.2.6-1973 and/or ASME Section XI, 1983 Edition through Summer 1983 Addenda, as applicable.

NDE Examinations

All ISI nondestructive examinations, including evaluation of flaw indications, were performed in accordance with the requirements stipulated under Section XI Subarticle IWA-2200, "Examination Methods".

The Inservice Inspections were performed from May 28, 1990 thru November 26, 1991. The Authorized Nuclear Inservice Inspector was Mr. Librado Malabanan from Hartford Steam Boiler Inspection and Insurance Company of Hartford, Connecticut, Chicago Branch, whose address is 2443 Warrenville Road, Suite 500 Lisle, Illinois 60532.

EXAMINATION SUMMARY

The following is a summary of the examinations performed during the Braidwood Unit 2 second refueling outage. Refer to the summary tables for additional information on specific welds, components, supports and pressure test examinations and their respective results.

PIPING SYSTEMS

CHEMICAL AND VOLUME CONTROL (CV)

A total of 5 pressure retaining welds were examined. Of these 5 received a surface exam (B10.10). One valve received a VT-1 exam (B7.70)

FEEDWATER (FW)

A total of 30 pressure retaining welds were examined. Of these received a volumetric and surface exam (C5.21) and 2 received surface exam only (C3.20). (see note 5 & 8)

MAINSTEAM (MS)

A total of 37 weldments were examined. Of these 17 received a surface exam (12-C3.20, 5-C5.31), 18 received both a volumetric and surface exam (11-C5.22, 7-C5.21) and 2 received a volumetric exam only (note 5)

REACTOR COOLANT (RC)

A total of 38 components were examined. Of these 12 welds were examined by surface & volumetric (10-B9.11, 2-B 5.70), 24 welds received a surface exam (6-B9.32, 4-B9.21, 12-B9.40, 2-B10.10), and 2 bolting connections received a VT-1 (B7.50) (see note 7)

RESIDUAL HEAT REMOVAL (RH)

A total of 3 pressure retaining welds received a surface exam (2-C5.12, 1-C5.11)

SAFETY INJECTION (SI)

A total of 10 pressure retaining welds were examined. One weld received a volumetric and surface exam (C5.21), and 9 welds received a surface exam (7-B9.40, 1-B10.10, 7-B9.40)

EMERGENCY CORE COOLING SYSTEMS

Five safety injection and 5 chemical and volume control weldments received a volumetric examination (see note 1)

COMPONENTS

REACTOR VESSEL (2RC01R)

A volumetric exam was performed from 0 to 180 degrees on the reactor vessel shell-to-flange weld (B1.30) and the threads in the flange (B6.40)

REACTOR COOLANT PUMPS

One (B7.60) bolting assembly was visually examined (VT-1), 4 reactor coolant pump flywheels were examined per Reg Guide 1.14 (see note 6).

PRESSURIZER (2RY01S)

One VT-1 was performed on the primary manway (B7.20)

RESIDUAL HEAT REMOVAL HEAT EXCHANGER (2RH02AA, 2RH02AB)

Four nozzle-to-shell (C-B C2.21) pressure retaining welds received a volumetric exam and 2 welds received a surface exam (see note 4)

RESIDUAL HEAT REMOVAL PUMP

3 pump lug welds (C-C C3.30) received a surface examination (see note 2)

STEAM GENERATOR (2RC01BA)

One VT-1 was performed on the primary manway (B7.30)

STEAM GENERATORS (2RC01BA, 2RC01BB, 2RC01BC, 2RC01BD)

Eddy Current examination was performed on all 4 steam generators. Fifty percent of each generators tubes received 100% full length examination and the remaining tubes were examined from the hot leg tube and thru the top support plate on the cold leg side. Based on Eddy Current results 11 tubes were plugged.

Additional information concerning the steam generator eddy current inspection can be obtained from the report submitted to the Nuclear Regulatory Commission as required by Braidwood Station Technical Specifications 4.4.5.5.

NIS-2
OWNER'S REPORT FOR
REPAIRS AND REPLACEMENTS

<u>SYSTEM</u>		<u>NIS-2</u>	<u>TOTAL</u>	<u>TOTAL</u>	<u>PAGES</u>
Containment Spray	(CS)	1		1	
Chemical Volume Control	(CV)	9		9	
Feedwater	(FW)	4		5	
Main Steam	(MS)	1		1	
Reactor Coolant	(RC)	14		14	
Residual Heat Removal	(RH)	1		1	
Pressurizer	(RY)	1		1	
Steam Generator Blowdown	(SD)	2		2	
Safety Injection	(SI)	3		3	
		<u>33</u>		<u>33</u>	

WELD/COMPONENT EXAMINATION NOTES

1. These are augmented volumetric examinations on 7 1/2% of the circumferential welds within the ECCS systems (refer to Note 10 of the ISI Program Plan)
2. Pump lug obstructions on the RHR pump (refer to NR-15 of the ISI Program Plan)
3. Braidwood Station will utilize Code Case N-426 (refer to Note 3 of the ISI Program Plan)
4. Indications exceeding Section XI acceptance criteria were detected on all (4) nozzle-to-vessel welds. Fracture mechanics analysis was performed and found acceptable. (ref. NRC Docket No. STN 50-457)
5. Ultrasonic examinations were performed on high energy lines not postulated for line breaks (refer to Note 5 of the ISI Program Plan)
6. Augmented NDE exams were performed on 4 reactor coolant pump flywheels (refer to Note 6 of the ISI Program Plan)
7. Only 25% thru wall indications can be detected on cast stainless steel welds (refer to Relief Request NR-2 of the ISI Program Plan)
8. A fabrication defect was detected on the 90 degree elbow adjacent to weld number 2FW-02-22 which exceeded the ASME Section XI ultrasonic testing criteria. The elbow was cut out, the indication removed and rewelded back into the system. The indication in the Main Feedwater elbow was not service induced therefore no sample expansion was performed.

Note:
Code Violation

Braidwood 2

**NDE ENGINEERING ANALYSIS & APPLICATIONS
INEL/EG&G IDAHO
IDAHO FALLS, IDAHO 83415-2209**

Telecopier Telephone: 208/526-0690
Verification Telephone: 208/526-6122

DATE: February 25, 1993

TO: Winston Liu, NRC RES

FAX: 301 492 3696

FROM: Jeff Cook, Sr.

TELEPHONE: 208 526 6414 office
208 522 7339 home

ASME Section XI Code Violation #2, Task 24

Per our telecon the following is provided for you to discuss with NRR. The licensee is using the 83-S83 Section XI. The violation was noted during review of the examination summary notes page. Both IWB-2430 and IWC-2430 require additional examinations when rejectable indications are revealed, and say nothing about exempting fabrication defects.

Number of Pages (including cover page): 3

C-132

INSERVICE EXAMINATION OF SELECTED CLASS 1, CLASS 2, AND CLASS 3 COMPONENTS AT THE CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 1

VOLUME I FINAL REPORT WITH APPENDICES SwRI Project 4863

Prepared for

**Baltimore Gas and Electric Company
Materials Engineering - Nuclear Unit
Calvert Cliffs Nuclear Power Plant
Lusby, Maryland 20657**

November 1992

Written by

**William A. Jensen
Project Engineer
NDE Engineering Section
Department of NDE Services**

Approved by

**David F. Rosoy
Director
Department of NDE Services
Nondestructive Evaluation Science
and Technology Division**

ABSTRACT

An inservice examination (ISI) of selected Class 1, Class 2, and Class 3 components of Baltimore Gas and Electric Company's (BG&E) Calvert Cliffs Nuclear Power Plant, Unit 1, was performed by BG&E and Southwest Research Institute (SwRI) personnel during the 1992 refueling outage. The ISI was performed using visual, manual ultrasonic, magnetic particle, and liquid penetrant techniques. This examination constituted the second ISI performed in the second 10-year interval of commercial operation. The components were examined in accordance with SwRI "Examination Plan for the 1992 Inservice Examination of Calvert Cliffs Nuclear Power Plant, Unit 1," including any changes made during the ISI as approved by BG&E.

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ABBREVIATIONS

ASME	-	American Society of Mechanical Engineers
BG&E	-	Baltimore Gas and Electric Company
ICN	-	Interim Change Notice
ISI	-	Inservice Examination
Calvert Cliffs 1	-	Calvert Cliffs Nuclear Power Plant, Unit 1
MT	-	Magnetic Particle
NDE	-	Nondestructive Examination
NPOP	-	Nuclear Projects Operating Procedure
NQAPM	-	Nuclear Quality Assurance Program Manual
PT	-	Liquid Penetrant
QA	-	Quality Assurance
RCP	-	Reactor Coolant Pump
RPV	-	Reactor Pressure Vessel
SSI	-	Sonic Systems International, Inc.
SwRI	-	Southwest Research Institute
UT	-	Ultrasonic
VT	-	Visual

I. INTRODUCTION

During the spring 1992 refueling outage, Southwest Research Institute (SwRI) and Baltimore Gas and Electric Company (BG&E) personnel performed nondestructive examinations (NDE) on selected Class 1, Class 2, and Class 3 components and their supports at BG&E's Calvert Cliffs Nuclear Power Plant, Unit 1 (Calvert Cliffs 1). This report discusses the results of those examinations.

A. Applicable Documents

The inservice examination (ISI) was performed in accordance with the following documents:

- Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1983 Edition with Addenda through Summer 1983 (83S83)
- Section V of the ASME Code, "Nondestructive Examination," 83S83
- SwRI document, "Program Plan for the Second Inspection Interval for Calvert Cliffs Nuclear Power Plant, Units 1 and 2"
- SwRI document, "Long-Term Plan for Calvert Cliffs Nuclear Power Plant, Unit 1"
- SwRI Final Plan, "Examination Plan for the 1992 Inservice Examination of Calvert Cliffs Nuclear Power Plant, Unit 1," with changes made on site and approved by BG&E personnel
- ASME Code Case N-307-1, "Ultrasonic Examination of Class 1 Bolting from Center-Drilled Holes"
- ASME Code Case N-408, "Alternative Rules for Examination of Class 2 Piping"
- SwRI Nuclear Quality Assurance Program Manual (NQAPM), Revision 3, with all applicable changes
- BG&E Program 5, Nondestructive Examination

B. Examination Areas

Selected components in the following Class 1, Class 2, and Class 3 areas were examined by SwRI and BG&E personnel utilizing NDE techniques:

Class 1 Components and Systems

Reactor Pressure Vessel

Upper Guide Structure
Core Barrel (In Place)
Flange Seal Surface and Ledge Area
Closure Head Nuts
Closure Head Surface-to-Expansion Compensating Ring
Closure Head Alignment Keyways

Piping

Shutdown Cooling
Safety Injection
Pressurizer Spray
Pressurizer Safety and Relief
Letdown Lines
Charging Lines
Drain Lines

Vessels

Pressurizer Manway - Studs and Nuts
Steam Generator No. 11 Manway - Studs and Nuts
Steam Generator No. 12 Manway - Studs and Nuts

Pumps

Reactor Coolant Pump Studs (Baseline)
Reactor Coolant Pump Flywheel
Reactor Coolant Pump Cap Screws

Valves

Bodies - Groups 2 and 4
Bolting

Class 2 Components and Systems

Piping

Safety Injection
Feedwater (Baseline)
Main Steam System

Pressure Retaining Components

Vessels
Piping
Pumps
Valves

Class 3 Components and Systems

Auxiliary Feedwater
Service Water
Component Cooling
Salt Water
Pressure Retaining

II. DISCUSSION OF EXAMINATION ACTIVITIES

A. Pre-Outage Planning

In order to assure efficient performance of the ISI at Calvert Cliffs 1, SwRI devoted a concentrated effort toward planning and preparing for the examination activities. This included preparation and review of nuclear projects operating procedures (NPOPs), review of ultrasonic (UT) calibration blocks for compliance to ASME Code and procedure requirements, and developing an Examination Plan. Such planning activities were conducted by SwRI's NDE Applications and NDE Engineering Sections with assistance from the Quality Assurance (QA) organization.

The Examination Plan was developed in one comprehensive document to provide all the technical information required to perform the ISI. It provided a listing of the examination areas, procedures, UT calibration block drawings, weld identification drawings, and other information pertinent to the performance of the ISI. Selection of Class 1, Class 2, and Class 3 components for examination was based on the Long-Term Plan for Calvert Cliffs 1. Additional components were scheduled for examination at the request of BG&E.

Communication was maintained between BG&E and SwRI engineers to review examination areas, discuss site support requirements, and develop the Examination Plan.

B. Examination Procedures and Personnel

The NDE activities were performed utilizing visual (VT), liquid penetrant (PT), magnetic particle (MT), and UT techniques. The reactor coolant pump flywheel (RCP11A) UT, all of the PT, and a large percentage of the VT and MT examinations were performed by BG&E personnel in accordance with BG&E procedures. These procedures were written and controlled in accordance with BG&E's QA program and are on file with BG&E.

The UT (except RCP11A flywheel), remote VT, and a portion of the VT and MT examinations were performed by SwRI personnel in accordance with SwRI procedures. The VT procedure and the UT procedure for the examination of studs were amended for specific examination purposes with Interim Change Notices (ICNs). Where an ICN was used, it was noted on the applicable individual examination data sheets and NDE Summary Table. Copies of the ICNs are included with the applicable procedures in Appendix C.

SwRI examination personnel were certified and qualified by SwRI in accordance with the requirements of SwRI NPOP 2.0-NDES-101, "Nondestructive Examination Personnel Qualification and Certification." Five Sonic Systems International, Inc. (SSI), personnel were employed by SwRI to assist in the performance of the ISI. They were under the control of SwRI and were considered a part of the SwRI examination team during this ISI at Calvert Cliffs 1. SSI personnel were certified, trained, and qualified by SSI in accordance with SSI's QA program. Copies of the SwRI and SSI examination personnel certifications are included in Appendix E. BG&E examination personnel were certified, trained, and qualified in accordance with BG&E's QA program. Copies of the BG&E examination personnel certifications are on file with BG&E.

C. Examination Activities

This section provides a discussion of the various onsite NDE activities performed during the ISI. Data records for examinations performed by BG&E are included in Appendix G. SwRI examination data records are included in Appendix H. A discussion of the equipment used for these examinations appears in Subsection F (Page 5) of this volume. While performing examinations, SwRI and BG&E personnel were subject to QA surveillance by BG&E QA personnel. The Authorized Nuclear Insurance Inspector and a representative of the United States Nuclear Regulatory Commission also performed surveillance of certain examinations performed by SwRI and BG&E personnel.

1. Reactor Pressure Vessel

SwRI Level II and Level III examiners used remote VT techniques to examine the interior of the reactor pressure vessel (RPV) and recorded the examinations on videotape. The interior portions of the RPV closure head were examined utilizing direct VT techniques. The RPV closure head nuts were examined utilizing an alternating current yoke and fluorescent MT material.

2. Class 1 and Class 2 Piping, Pumps, and Valves

SwRI and BG&E Level II and Level III examiners used MT, PT, and VT techniques to examine Class 1 and Class 2 components, including austenitic piping, ferritic piping, dissimilar metal welds, flange bolting, valve bolting, and pumps. Various techniques were utilized by SwRI personnel to perform the manual UT examinations, dependent upon the material type and weld thickness. For a typical piping weld, the following techniques were used:

- a. If this was the first examination of the weld, a 0-degree lamination scan (UT0L) was performed to detect any laminar reflectors which may have affected the interpretation of angle-beam results. Also, an outside surface weld contour and accompanying thickness measurements were obtained and documented.
- b. For austenitic and ferritic piping welds, angle-beam scans of 45 degrees (UT45) were used for the detection of reflectors oriented parallel to the weld.
- c. Angle-beam scans, using 45-degree search units directed parallel to the weld (UT45T) were used on piping for the detection of reflectors oriented transverse to the weld. In ferritic piping where previous UT45T examinations had revealed no recordable indications, this examination is not required by ASME Section XI Code or ASME Code Case N-408.
- d. In austenitic piping welds, angle-beam scans using 45-degree search units directed tangentially to the weld (UT45TAN) were used for the detection of reflectors branching from a transverse or circumferential flaw.

3. Bolting

The reactor coolant pump (RCP) bolting was examined in accordance with ASME Code Case N-307-1.

- (a) For the baseline UT examinations of four RCP studs, a UT0, UT50, and UT70 were used to provide complete coverage of the examination area.
- (b) A UT88 surface wave was utilized for detection of reflectors propagating from the RCP stud heater holes.

D. Summary of Examination Results

UT indications determined to be recordable were investigated by SwRI to determine the nature of the reflector. [REDACTED]

During the remote VT examination of the reactor vessel, weld-like indications, tool marks, scratches, and debris were noted. All indications, scratches, and tool marks were determined to be nonrelevant. The debris was recorded on Issue Report IRO-003-298 and subsequently removed by BG&E personnel. In addition to the videotape recordings, further details of the VT examination areas and results are recorded in Engineering Test Procedure 86-24 in Appendix C.

Details of the examinations performed by BG&E personnel are presented in ~~BG&E NDE data records~~. Copies of these data records are contained in Appendix G.

E. Data Comparison

In accordance with IWB-3121 of Section XI, examination data were compared with recorded results of the preservice examination and/or prior ISIs. When the comparison was performed by SwRI, the examination records were annotated with the following statement: "Results have been compared to previous data." Comparisons made by BG&E are listed in Report No. 92-50-194 located in Volume II.

F. Equipment

Various NDE equipment was used during the ISI to perform examinations of the selected components. The major equipment items are discussed below.

1. Sonic FTS MK I and Sonic 136 Ultrascopes

Sonic FTS MK I and Sonic 136 UT instruments were used for manual UT examinations and thickness measurements of materials.

To assure proper instrument linearity and operation, SwRI certified each SwRI-owned Sonic prior to use at Calvert Cliffs 1, as specified in the NQAPM. Copies of certifications for SwRI-owned instruments utilized at Calvert Cliffs 1 are contained in Appendix F. Additionally, instrument linearity was checked and documented by SwRI personnel on site prior to and after the performance of the manual UT examinations.

2. Transducers

Various brands, sizes, types, and frequencies of UT transducers were used to perform the manual UT examinations at Calvert Cliffs 1. A transducer frequency of 1.5, 2.25, or 5.0 MHz was used as specified in the applicable NPOP.

Prior to use at Calvert Cliffs 1, each SwRI-owned transducer was tested and certified to be within SwRI's acceptance standards. This certification is performed within 12 months of use to verify that each transducer is still performing within the specified limits. A copy of each SwRI-owned transducer's certification is contained in Appendix F.

Information on the actual transducer utilized for any specific examination may be determined by reviewing the copies of the data sheets and referenced calibration sheets in Appendix H.

5. Stud Probe

Baseline UT examinations of four RCP studs were performed using the SwRI-designed Stud Probe (see Figure 1). The Stud Probe is inserted into the heater hole of the stud and is moved up and down by the examiner with a slight rotation after each pass. This process is continued until 100 percent of the accessible volume of the stud is examined.

6. Magnetic Particle Examination Equipment

MT examinations were performed utilizing hand-held alternating current magnetic particle yokes. In the case of the RPV closure nuts, fluorescent MT spray and blacklights were used with the yokes to perform the examinations. Prior to and after each examination, the

blacklight and yoke characteristics were checked by SwRI examiners and demonstrated to be satisfactory in accordance with the applicable NPOPs. All equipment and materials utilized by SwRI examiners for the MT examinations were provided by BG&E. All appropriate certifications are on file with BG&E.

Figure 1. Conceptual Drawing Showing the Operation of the Manual Stud Probe Device

G. Materials

In addition to the equipment previously discussed, certain materials were required in the performance of the examinations. All materials contacting the examination surface (i.e., ultrasonic couplant, piping marking devices, liquid penetrant, etc.) are required by the ASME Code to be tested and certified to be within acceptable sulfur and halogen limits prior to their use. Certifications for materials used by SwRI for the performance of examinations are contained in Appendix F. ~~Certifications for materials supplied by BG&E for the performance of examinations are on file with BG&E.~~

H. Ultrasonic Calibration Blocks

Pipe calibration blocks owned by BG&E were used to calibrate manual UT instruments prior to the examination of the selected welds. Drawings of the various calibration blocks used are contained in Appendix D. Certifications for the blocks are on file with BG&E.

III. SUMMARY OF EXAMINATIONS

This section provides a discussion of the field data records and a summary of the NDE activities performed at Calvert Cliffs 1.

A. Explanation of Field Data Records

The results of the examinations and calibrations performed by SwRI personnel were recorded on standard SwRI forms. The results of the examinations performed by BG&E personnel were recorded on either SwRI or BG&E forms. Copies of the completed SwRI documents constitute Appendix H of this report. Completed BG&E forms are included in Appendix G. The original SwRI records will be retained in the SwRI Records Vault. The original BG&E records will be retained by BG&E.

The field data records for each weld or area are assembled in a package preceded by an SwRI Examination Summary Record (summary sheet). The examination areas and summary sheet numbers correspond to those listed in the Summary Table. A general explanation of the individual field data forms follows.

The instruments utilized to perform the UT examinations were calibrated prior to use, then verified again at intervals specified in the applicable NPOPs, and upon the completion of the examinations. The calibration parameters were recorded prior to examination on the appropriate calibration record sheet as specified by the applicable NPOP. The documented calibration and calibration verification provide immediate assurance that the examinations were performed using properly calibrated instruments.

The examination results were recorded on the applicable data record sheets as specified in the applicable NPOP. The information documented on these forms describes the parameters associated with those indications which were greater than the recording levels specified in the applicable NPOP.

The size, location, and nature of indications were determined by analyzing the indication parameters recorded on the forms described above. The analysis is documented on SwRI Indication Resolution Record Sheets, which are included as part of each data package, when required.

B. Nondestructive Examination Summary Table

The following section is the Nondestructive Examination Summary Table (Summary Table). The Summary Table lists the areas at Calvert Cliffs 1 that were examined by SwRI or BG&E during this ISI. See Figure 2 for an explanation of the Summary Table format. Weld identification drawings depicting the location of RPV and Class 1 piping components are contained in Appendix A. Class 2 piping components are contained in Appendix B.

DATE:
REVISION:

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, FIRST PERIOD, SECOND OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE:

REACTOR PRESSURE VESSEL

SUMMARY EXAMINATION AREA		ASME	EXAM	PROCEDURE	N O O G T R E H E O E C M R	REMARKS
NUMBER	IDENTIFICATION	SEC. XI CATGY ITEM NO	METHOD			**CALIBRATION BLOCK**
						<p>The remarks column is used to describe any pertinent or unique features of the examination such as limitations, results, CNFs, etc. Ultrasonic calibration blocks are also listed in this column.</p>
						<p>The results of the examination are indicated in these columns. The absence of recordable indications is shown by an "X" in the "NOREC" column. The presence of ultrasonic indications shown to be the result of a geometric feature of the examination area is indicated by an "X" in the "GEOM" column. The presence of nongeometric indications is indicated by an "X" in the "OTHER" column, and an explanation of the nature of each nongeometric indication is contained in the "REMARKS" column.</p>
						<p>This column lists the applicable SWRI nuclear projects operating procedure used for the examination.</p>
						<p>The NDE method used during the examination is listed in this column.</p>
						<p>The ASME SECTION XI ITEM NO. and CATEGORY of the examination area are listed in this column.</p>
						<p>Each examination area is listed in this column. Details of the weld identification system are contained in Appendices A and B.</p>
						<p>This column references the examination summary sheet which serves as a cover sheet for the data package and lists the data record numbers, the examiners, and any pertinent remarks.</p>

Figure 2. Explanation of the Summary Table Format

DATE: 11/05/92
REVISION: 0

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, FIRST PERIOD, SECOND OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE

REACTOR PRESSURE VESSEL

					N	O	
		ASME			O	G	T
		SEC. XI			R	E	N
SUMMARY EXAMINATION AREA		CATGY	EXAM		E	O	E
NUMBER IDENTIFICATION		ITEM NO	METHOD	PROCEDURE	C	M	R
.....					-	-	-

VESSEL INTERNALS (FIG NO A-1)

003300	VESSEL INTERIOR	B-N-1	VT-1	ETP-86-24 Rev.1	X	-	-	CORE BARREL IN PLACE. FOUR (4) F
		B13.10	VT-3	CCN-VT7	X	-	-	WELD-LIKE AREAS NOTED AROUND THE
								DEGREE OUTLET NOZZLE. VARIOUS LI
								SCRATCHES AROUND THE OUTLET NOZZL
								SEE ETP REPORT 86-24 IN APPENDIX

003490	CORE SUPPORT STRUCTURES	B-N-3	VT-1	ETP-86-24 Rev.1	X	-	-	CORE BARREL IN PLACE. SEE ETP RE
		B13.70	VT-3	CCN-VT7	X	-	-	86-24 IN APPENDIX C.

NUTS (FIG NO A-1 & A-2)

003950	NUT NUMBERS 1 THROUGH 54	B-G-1	MT	BGE-MT2/O	X	-	-	EXAMINATION PERFORMED BY SWRI & B
		B6.10		5.107, REV. 2				PERSONNEL. EXAMINED NUT NUMBERS
								THROUGH 18. SEE NDE REPORT 92-50
								IN APPENDIX G.

DATE: 11/05/92
REVISION: 0

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, FIRST PERIOD, SECOND OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE:

STEAM GENERATOR NUMBER 11

SUMMARY EXAMINATION AREA	ASME				N	O	
NUMBER IDENTIFICATION	SEC. XI	CATGY	EXAM		O	G	T
		ITEM NO	METHOD	PROCEDURE	R	E	H
					E	O	E
					C	M	R
					.	.	.
							REMARKS
							CALIBRATION BLOCK

STUDS AND NUTS (FIG NO A-5)

005700	OUTLET SIDE MANWAY STUDS	B-G-2 87.30	VT-1	5.712, REV. O	X	.	.	EXAMINATION PERFORMED BY BG&E PER: SEE NDE REPORT 92-10-1257 IN APPEN
005730	OUTLET SIDE MANWAY NUTS	B-G-2 87.30	VT-1	5.712, REV. O	X	.	.	EXAMINATION PERFORMED BY BG&E PER: SEE NDE REPORT 92-10-1257 IN APPEN
005800	INLET SIDE MANWAY STUDS	B-G-2 87.30	VT-1	5.712, REV. O	X	.	.	EXAMINATION PERFORMED BY BG&E PER: SEE NDE REPORT 91-10-1257 IN APPEN
005850	INLET SIDE MANWAY NUTS	B-G-2 87.30	VT-1	5.712, REV. O	X	.	.	EXAMINATION PERFORMED BY BG&E PER: SEE NDE REPORT 92-10-1257 IN APPEN

DATE: 11/05/92
REVISION: 0

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, FIRST PERIOD, SECOND OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE: 12

STEAM GENERATOR NUMBER 12

SUMMARY EXAMINATION AREA		ASME			N	O	
NUMBER	IDENTIFICATION	SEC. XI	CATGY	EXAM		O	G T
		ITEM NO	METHOD	PROCEDURE		R	E H
						E	O E
						C	M R
							REMARKS
							CALIBRATION BLOCK
<u>STUDS AND NUTS (FIG NO A-5)</u>							
007000	OUTLET SIDE MANWAY STUDS	B-G-2 87.30	VT-1	5.712, REV. O	X	-	EXAMINATION PERFORMED BY BG&E PERSONNEL. SEE NDE REPORT 92-10-1256 IN APPENDIX G.
007050	OUTLET SIDE MANWAY NUTS	B-G-2 87.30	VT-1	5.712, REV. O	X	-	EXAMINATION PERFORMED BY BG&E PERSONNEL. SEE NDE REPORT 92-10-1256 IN APPENDIX G.
007100	INLET SIDE MANWAY STUDS	B-G-2 87.30	VT-1	5.712, REV. O	X	-	EXAMINATION PERFORMED BY BG&E PERSONNEL. SEE NDE REPORT 92-10-1256 IN APPENDIX G.
007150	INLET SIDE MANWAY NUTS	B-G-2 87.30	VT-1	5.712, REV. O	X	-	EXAMINATION PERFORMED BY BG&E PERSONNEL. SEE NDE REPORT 92-10-1256 IN APPENDIX G.

NIS-1

OWNER'S DATA REPORT FOR INSERVICE INSPECTION
(As Required by the Provisions of the ASME Code Rules)

1. Owner: Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203
2. Plant: Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702
3. Plant Unit: One (#1)
4. Owner Certification of Authorization: N/A
5. Commercial Service Date: May 8, 1975
6. National Board Number for Unit: 20911
7. Components Inspected:

COMPONENT OR APPURTENANCE	MANUFACTURER OR INSTALLER	MANUFACTURER OR INSTALLER SERIAL #	STATE OR PROVINCE #	NATIONAL BOARD NUMBER
Reactor Pressure Vessel	Combustion Engineering	CE-67107	14000NV	20911
Pressurizer	Combustion Engineering	CE-67602	14000NV	20915
Reactor Pressure Vessel Closure Head	Combustion Engineering	CE-67207	14000NV	20911
Steam Generator #11	Combustion Engineering	CE-67504	14000NV	20922
Steam Generator #12	Combustion Engineering	CE-67505	14000NV	20923
#12A Reactor Coolant Pump	Byron Jackson	SN-0439	14000NV	
#12B Reactor Coolant Pump	Byron Jackson	SN-0440	14000NV	
#11A RCP Motor Flywheel	Westinghouse	SN-2S-79P528		
Safety Injection Piping/Supports	Bechtel			

7. Components Inspected: (con't.)

COMPONENT OR APPURTENANCE	MANUFACTURER OR INSTALLER	MANUFACTURER OR INSTALLER SERIAL #	STATE OR PROVINCE #	NATIONAL BOARD NUMBER
Main Steam Piping	Bechtel			
Pressurizer Spray Piping	Bechtel			
Shutdown Cooling Piping/Supports	Bechtel			
Pressurizer Safety & Relief Piping	Bechtel			
Charging Piping	Bechtel			
Drain Piping	Bechtel			
Letdown Piping	Bechtel			
Auxiliary Feedwater Piping/Supports	Bechtel			
Service Water Piping/Supports	Bechtel			
Component Cooling Water Piping/Supports	Bechtel			
Salt Water Piping/Supports	Bechtel			

8. Examination Dates: July 5, 1988 to August 20, 1992

9. Inspection Interval: April 1, 1987 to April 1, 1998

10. Abstract of Examinations: (Include list of exams and statements concerning status of work required for current interval)

(See Enclosure 3)

The examinations reported herein constitute the tenth such report of Inservice examination performed at Calvert Cliffs Unit 1, and the second report within the first period of the second interval of commercial operation. The examinations for the second interval as required by the Long-Term Examination Plan for Calvert Cliffs Unit 1, were performed in accordance with the 1983 Edition of ASME Code Section XI with Addenda through Summer, 1983. Hydrostatic and System Pressure Testing Procedures are available at the plant site for review.

NIS-1

11. Abstract of Conditions Noted:

(See Enclosures 2 & 3)

- a. Eddy current examinations of #11 and #12 Steam Generators in 1992 revealed signs of steam generator tube degradation. Results of #11 Steam Generator examinations showed that of the 8,412 tubes examined, ten (10) were defective ($\geq 40\%$ wall loss), 112 were degraded ($\geq 20\%$ wall loss, but $< 40\%$), 162 had imperfections ($< 20\%$ wall loss). Inspection of #12 Steam Generator revealed that of the 8,449 tubes examined, five (5) were defective, 120 were degraded, 172 had imperfections. As a result of this inspection, eleven (11) tubes were plugged in #11 Steam Generator and seven (7) tubes were plugged in #12 Steam Generator.
- b. A remote visual examination of the Reactor Pressure Vessel (RPV) interior was performed. The RPV Core Barrel interior surfaces revealed scratches and tool marks above outlet nozzles. Additionally, several foreign materials were found inside the RPV on top of Core Plate.
- c. Visual examinations of supports revealed 15 supports with construction/installation deficiencies; 3 supports with missing or loosened items; 1 support with improper spring settings; and 7 supports with other deficiencies.
- d. Due to administrative and clerical problems, 63 NIS-2's are not included with this submittal.

12. Abstract of Corrective Measures Recommended and Taken:

(See Enclosure 3)

- a. Eleven (11) tubes were plugged in #11 Steam Generator and 7 tubes were plugged in #12 Steam Generator.
- b. The surface scratches and tool marks were resolved as nonrelevant indications. All foreign material was retrieved and removed.
- c. The supports were either accepted by evaluation in accordance with IWF-3122.4 or repaired to bring them into conformance with their original design.
- d. Issue reports have been generated to address the delayed NIS-2's. Corrective measures and actions to prevent recurrence will be determined in accordance with the Calvert Cliffs Issue Assessment System process. The outstanding NIS-2's will be submitted with the next Unit #1 Summary Report.

13. Names of Inspectors: R. W. Lawrence and N. S. Hewitt

Employer of Inspectors: Factory Mutual Engineering
401 City Avenue, Suite #420
Bala Cynwyd, PA 19004

We certify that the Statements made in this report are correct and the examinations and corrective measures taken conform to the rules of the ASME Code, Section XI.

DATE: November 17, 1992.

SIGNED:

R. B. Pond, Jr. For, Baltimore Gas & Electric Company
R. B. Pond, Jr. - Principal Metallurgist
Materials Engineering & Analysis Unit

Certificate of Authorization No.

N/A

Expiration Date

N/A

CALVERT CLIFFS NUCLEAR POWER PLANT

SUMMARY OF NIS-2 REPORTS

UNIT 1

DATE: 11/17/92

PAGE: 1

MO NUMBER	COMPONENT
18801458	Snubber 1-52-18
18806194	DR-5 Flow Oriface
18806322	1-CV-3939
18806594	Pipe
18806880	Check Valve
18807219	Snubber 1-52-9
18808008	Snubber 1-61-1
18808029	Snubber 1-52-10
18808032	Snubber 1-52-11
18808228	RCP Seal
18808424	Snubber 1-83-69
18808480	Reactor Coolant Pump
18808736	Valve (Main Plug)
18809812	Component Cooling system pipe support
18809889	Piping System
18809889	Piping System
1880989	Piping System
18900902	Charging Pump discharge line Support
18900902	Discharge Gate Valve
18901783	Snubber 1-52-13
18901786	Snubber 1-61-1
18902423	Discharge Desurger Supp.
18903720	Snubber 1-52-58

7 pages
41 non supports
out of 156

MO NUMBER	COMPONENT
18904343	CC-2 piping to LT-110Y & LT-103
18904344	CC-2 Piping to LT-110Y & LT-103
18904673	Snubber 1-52-22
18905056	Snubber 1-41-8
18905198	Snubber 1-64-12
18905376	Hanger rod on 2" EB-6-1008
18906134	Heat Exchanger
18907210	Pipe Support 14"GC-5-1004 R-2
18907241	BOTTOM BLOWDOWN CONTROL VALVE
18907269	Pressurizer
18907321	Snubber 1-52-57
19001282	Snubber 1-52-34
19002157	Snubber 1-41-4
19002160	Snubber 1-41-7
19002262	Snubber 1-45-7
19002267	Snubber 1-52-16
19002270	Snubber 1-52-21
19002368	Snubber 1-52-25A
19002371	Snubber 1-52-26
19002374	Snubber 1-52-32A
19002376	Snubber 1-52-36
19002391	Snubber 1-52-42A
19002393	Snubber 1-52-45
19002405	Snubber 1-60-18
19002422	Snubber 1-61-17
19002454	Snubber 1-63-28
19002470	Snubber 1-64-8
19002519	Snubber 1-52-42A

MO NUMBER	COMPONENT
19002589	Snubber 1-64-26
19003727	Piping
19004165	Snubber 1-52-36
19004237	Snubber 1-52-50A
19005335	Snubber 1-64-2
19005465	Snubber 1-64-20
19005469	Snubber 1-64-23
19005484	Snubber 1-64-11
19005530	Snubber 1-64-17
19005589	Snubber 1-52-42
19005642	Piping
19005642	Piping
19005664	Snubber 1-64-31
19009100	Piping
19009177	REfueling Water Tank
19009208	Snubber 1-63-17
19009265	Snubber 1-64-45
19009447	Support 10" GC-1-1003-H18
19009511	PIPE SUPPORT SK-1422
19009535	PIPE RESTRAINT 14" HC-3-1011-R7
19009548	6"GC-3-1006-H16
19009557	MS PIPING BETWEEN LINER PLATE & ISOL. VALVES
19009620	Support Rod 6" GC3-1006-H16
19009621	Rod for support piping 4" GC 3-1005-H53
19009650	REACTOR COOLANT PUMP SEAL (UPPER SEAL GLAND)
19009848	Valve

MO NUMBER	COMPONENT
19009851	Valve
19009873	Valve
19010229	Pressure Vessel
19100237	Snubber 1-60-25
19101234	Snubber 1-63-20
19101572	Snubber 1-63-27
19102799	Pressure Vessel
19102886	SNUBBER 1-41-5
19102887	SNUBBER 1-45-1A
19102888	Snubber 1-52-7
19102889	SNUBBER 1-52-46
19102890	SNUBBER 1-52-62
19102895	SNUBBER 1-50-26
19102912	SNUBBER 1-83-48
19102912	Snubber 1-83-48
19102913	Snubber 1-83-47
19102914	Snubber 1-64-16
19102918	SNUBBER 1-64-18
19102919	SNUBBER 1-64-97
19102920	SNUBBER 1-64-17A
19103186	Supports
19103230	Pressure Vessel
19103230	Pressure Vessel
19103230	Pressure Vessel
19103230	Pressure Vessel
19103230	Pressure Vessel
19103230	Pressure Vessel
19103475	SNUBBER 1-64-32

MO NUMBER	COMPONENT
19103653	Snubber 1-83-69A
19103716	SNUBBER 1-63-13
19103727	SNUBBER 1-63-18
19103786	Snubber 1-63-19
19103815	SNUBBER 1-41-3
19103816	SNUBBER 1-52-29
19103819	SNUBBER 1-63-9
19103881	Snubber 1-52-6
19104544	Valve
19105344	Snubber 1-52-24
19105347	Snubber 1-41-15
19105348	Snubber 1-52-25B
19105349	Snubber 1-52-28A
19105351	Snubber 1-52-32
19105352	Snubber 1-52-38
19105353	Snubber 1-52-43A
19105354	Snubber 1-52-46A
19105355	Snubber 1-52-57
19105356	Snubber 1-52-58A
19105357	Snubber 1-52-59
19105359	Snubber 1-52-60
19105366	Snubber 1-63-10
19105368	Snubber 1-63-14
19105369	Snubber 1-63-15
19105370	Snubber 1-63-19
19105371	Snubber 1-63-25
19105372	Snubber 1-63-26
19105373	Snubber 1-64-7

MO NUMBER	COMPONENT
19105374	Snubber 1-64-10
19105375	Snubber 1-64-15
19105376	Snubber 1-52-39
19105377	Snubber 1-64-35
19105379	Snubber 1-64-99
19105380	Snubber 1-64-100
19105383	Snubber 1-83-4
19105387	Snubber 1-83-70
19105538	Snubber 1-41-12
19105539	Snubber 1-52-3
19105540	Snubber 1-52-43
19105541	Snubber 1-52-48
19105542	Snubber 1-52-15
19105559	Valve
19105566	Valve
19105829	Snubber 1-52-13A
19106531	Snubber 1-45-7
19106704	Snubber 1-60-20
19106705	Snubber 1-60-27
19106909	Check Valve 1-SI-245
19107001	Snubber 1-52-17
19107002	Snubber 1-52-74
19107351	Valve
19107351	Valve
19200418	Reactor Coolant Piping
19200449	Pressure Vessel
19200483	Valve
19200487	Snubber 1-64-96

NO NUMBER	COMPONENT
19200490	Snubber 1-52-56
19201268	Pressure Vessel
19201269	Pressure Vessel
19201885	Snubber 1-52-39
19201889	Snubber 1-64-12
19202923	Snubber 1-52-30
19202998	Reactor Coolant Piping
19203529	Piping System
19203529	Piping System
19203531	Piping System
19203638	Valve
19203929	Snubber 1-45-2
19203951	Valve
19204793	Snubber 1-64-97

C-160

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MS-2209

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U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Attn: A. B. Davis

January 22, 1993

Dear Mr. Davis:

Attached please find a copy of the ISI NIS-1 Report for Inservice Inspection activities performed at the Donald C. Cook Nuclear Power Plant, Unit 1, State of Michigan, Number M-09672-M, located in Bridgman, Michigan. The Plant is owned by Indiana Michigan Power Company, One Summit Square, Ft. Wayne, Indiana 46802.

The unit's commercial service date is August 23, 1975 and has a gross generating capacity of 1080 MWe.

The Authorized Code Inspector was Mr. M. K. Muterspaugh from Factory Mutual Engineering, whose address is 30150 Telegraph Road, Bingham Office Park Suite 141, Bingham Farms, Michigan 48010-4250.

The examinations, tests, replacements and repairs performed, conditions observed, and corrective measures taken are summarized in the attached report. ISI activities were performed in accordance with the rules and requirements of ASME Code Section XI 1983 Edition, Summer of 1983 Addenda.

Respectively,

J. E. Rutkowski
Assistant Plant Manager
D. C. Cook Nuclear Plant

/cbm

Attachment

c: File

250136

9301270178 930122
PDR ADOCK 05000276
G PDR

C-162

AC47

INDIANA
MICHIGAN
POWER

DONALD C. COOK
NUCLEAR PLANT

UNIT 1

REPORT OF INSERVICE INSPECTIONS

FROM JANUARY 27, 1991
THROUGH OCTOBER 28, 1992

9301270180 930122
PDR ADOCK 05000278
G PDR

C-163

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

REPORT OF INSERVICE INSPECTIONS
FROM JANUARY 27, 1991 TO OCTOBER 28, 1992

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Section 1	Introduction
Section 2	NIS-1 Inservice Inspection Summary Report
Section 3	Examination of Class 1 and Class 2 Systems and Components
Section 4	Steam Generator Eddy Current Examinations, Tube Plugging and Tube Sleeving.
Section 5	System Pressure Tests
Section 6	NIS-2 Repair and Replacement Reports

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

INTRODUCTION

This report summarizes the Inservice Inspection activities performed at the Cook Nuclear Plant, Unit 1 for the period of January 27, 1991 through October 28, 1992. These activities were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. As of July 1, 1986, the second ten-year inspection interval began and the applicable code for ISI activities then became the 1983 edition through the summer 1983 addenda.

The purpose of this report is to satisfy the requirements of IWA-6000 for the submittal of an Inservice Inspection summary report at the completion of each refueling outage. Examinations and tests conducted since the preceding summary report (January 26, 1991) are included in this report. Repairs and replacements are also included in this report per the requirements of the 1983 Edition of ASME Section XI IWA-6220(b)10. Also, contained within this report are the supplemental information requirements of IWA-6220.

Contents of this report have been reviewed and verified correct with supporting documentation available at the plant for review if required.

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

EXAMINATION OF CLASS 1 AND CLASS 2 COMPONENTS

Inservice examinations were performed during the 1992 Unit 1 Refueling Outage during the period of January 27, 1991 through October 28, 1992. All examinations were in accordance with ASME Section XI, 1983 Edition through 1983 Summer Addenda.

Examinations were scheduled and performed according to the Second Ten-Year Interval Long-Term Inservice Examination and Testing Plan for Class 1, 2 and 3 Systems and Components for Donald C. Cook Nuclear Plant, Unit 1. This is the currently approved ISI examination plan, which identifies specific activities to be performed on a refueling outage basis.

Nondestructive examinations were performed by Southwest Research Institute (SWRI), MQS Inspections and Indiana Michigan Power (IMP) personnel in accordance with procedures which were approved for use at the Cook Nuclear Plant. SWRI performed the required volumetric ultrasonic examinations and surface examination if one was needed in addition to the volumetric examination. MQS and IMP personnel performed all other surface and visual examinations except for the reactor vessel visual examination which was performed by Westinghouse.

A reactor vessel visual examination was performed of areas accessible during a normal refueling outage. In addition, the upper internals and accessible areas of the lower internals was visually examined. No recordable indications were noted.

Snubber examinations and functional testing were performed in conjunction with maintenance procedures for technical specification surveillance of snubbers.

IMP personnel performed VT-3 and VT-4 visual examinations on ISI component supports. Design sketch nonconformances and other discrepancies found were evaluated by AEPSC Structural and Analytical Design Section. All discrepant pipe supports and associated systems were found to be functional and met operability limits. However, some of the pipe supports were repaired to be in accordance with the original design intent.

The summary pages in this section lists the examinations, tests performed and results. In the following summary pages, "NOREC" is no recordable indications, "GEOM" is geometric indications, and "OTHER" are indications other than geometric indications.

INDIANA MICHIGAN POWER COMPANY
D.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975

DATE: 01/07/88 NATIONAL BOARD NUMBER 20761 COOK NUCLEAR PLANT UNIT 1

PAGE: 2

REVISION: 0

SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
OUTAGE 2 (1992), SECOND PERIOD, SECOND INTERVAL
CLASS 1 COMPONENTS

REACTOR COOLANT SYSTEM

SUMMARY EXAMINATION AREA		ASME			N	O	
NUMBER	IDENTIFICATION	SEC. XI			O	G	T
		CATGY	EXAM		R	E	H
		ITEM NO	METHOD	PROCEDURE	E	O	E
					C	M	R
							REMARKS
							CALIBRATION BLOCK

<u>LINE 1-RC-10 (FIG NO A-15)</u>							
038400	1-RC-10-07S	B-J	UT45	DCC-UT31/1/0/	X	-	SWRI - AUGMENTED EXAM FOR NRCB 88-08.
	PIPE TO TEE	89.11	UT45T	1 & 3	X	-	
			UT60		X	-	
			UT60T		X	-	
							3378027 (4-SS-120-.430)
038500	1-RC-10-08F	B-J	UT45	DCC-UT31/1/0/	X	-	SWRI - AUGMENTED EXAM FOR NRCB 88-08.
	TEE TO PIPE	89.11	UT45T	1 & 3	X	-	
			UT60		X	-	
			UT60T		X	-	
							3378027 (4-SS-120-.430)
038600	1-RC-10-09F	B-J	UT45	DCC-UT31/1/0/	X	-	SWRI - AUGMENTED EXAM FOR NRCB 88-08.
	TEE TO PIPE	89.11	UT45T	1 & 3	X	-	
			UT60		X	-	
			UT60T		X	-	
							3378027 (4-SS-120-.430)
038700	1-RC-10-10S	B-J	UT45	DCC-UT31/1/0/	X	-	SWRI - AUGMENTED EXAM FOR NRCB 88-08.
	PIPE TO ELBOW	89.11	UT45T	1 & 3	X	-	
			UT60		X	-	
			UT60T		X	-	
							3378027 (4-SS-120-.430)

COOK NUCLEAR PLANT UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
OUTAGE 2 (1992), SECOND PERIOD, SECOND INTERVAL
CLASS 1 COMPONENTS

PAGE :

SAFETY INSPECTION STATUS				N	O	REMARKS	
				O	G		
				R	E		
				E	O		
SUMMARY	EXAMINATION AREA	ASME	EXAM	C	M	R	
NUMBER	IDENTIFICATION	SEC. XI	CATGY	ITEM NO	METHOD	PROCEDURE	**CALIBRATION BLOCK**
<u>LINE 1-SI-23 (FIG NO A-33)</u>							
093500	1-SI-23-05F COLLAR TO PIPE	B-K-1 B10.10	PT UTOL UT45	12SHP5050NDE001 DCC-UT49/1/0/1 DCC-UT31/1/0/3	- - X X - - X - -	J.O. C412 - PT LINEAR INDICATION 9/16" LONG. SWRI - LINEAR INDICATION EXAMINE BY UT METHOD IN ACCORDANCE WITH IWB-3516.1 PARA. (C) AND FOUND TO BE ACCEPTABLE.	
<u>LINE 1-SI-546 (FIG NO A-42A)</u>							
117600	1-SI-546-31S COLLAR TO PIPE	B-K-1 B10.10	PT	12SHP5050NDE001	X - -	J.O. C412 - LIMITED EXAM AREA DUE TO BOLTING.	

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
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Technical Specification/Inservice Inspection
Surveillance of Hydraulic Snubbers

INDIANA MICHIGAN POWER COMPANY
C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

TS/ISI SNUBBER EXAMINATION AND TESTING

Snubber visual examinations are separated into three (3) categories: Grinnell "Accessible", Grinnell "Inaccessible", and "Steam Generator". The functional test surveillances are separated into two (2) categories: "Grinnell" and "Steam Generator".

Lists of those snubbers classified as "Accessible", "Inaccessible", and "Steam Generator" are attached.

Visual Examinations

<u>Category</u>	<u>Compl. Date</u>	<u>Comments/Results</u>
a.Grinnell "Accessible"	06-16-92	all acceptable
	09-25-92	all acceptable
b.Grinnell "Inaccessible"	08-27-92	all acceptable
	10-03-92	all acceptable
c.Steam Generator	07-18-92	all acceptable
	09-19-92	all acceptable

Functional Testing

<u>Category</u>	<u>Compl. Date</u>	<u>Comments/Results</u>
a.Grinnell	09-28-92	all acceptable, see attached sheet
b.Steam Generator	08-11-92	three tested, all acceptable, see attached sheet

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MI.
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

UNIT 1 STEAM GENERATOR SNUBBERS

<u>STEAM GENERATOR NO.</u>	<u>TECH. SPEC. NO.</u>	<u>SERIAL NUMBER</u>
S/G #11	79	25.12620.007-1
	80	25.12620.007-5
	81	25.12620.007-7
	82	25.12620.007-14

S/G #12	85	25.12620.007-9
	84	25.12620.007-6
	86	25.12620.007-15
	83	25.12620.007-3

S/G #13	88	25.12620.007-4
	87	25.12620.007-2
	90	25.12620.007-16
	89	25.12620.007-8

S/G #14	93	25.12620.007-12
	92	25.12620.007-11
	94	25.12620.007-13
	91	25.12620.007-10

MICHIGAN POWER COMPANY
 C. COOK NUCLEAR PLANT, BRIDGMAN, MI
 UNIT 1 COMMERCIAL SERVICE DATE 8-23-1977
 NATIONAL BOARD NUMBER 20761

UNIT 1 SNUBBERS

FUNCTIONAL TEST SURVEILLANCES

<u>Snubber Mark No.</u>	<u>Date Removed</u>	<u>Date Reinstalled</u>	<u>As-Found</u>
1-GRC-S564	08-21-92	08-27-92	Pass
1-GRC-S573	08-21-92	08-27-92	Pass
1-GRC-S592	08-18-92	08-26-92	Pass
*1-GRC-S614	08-22-92	09-08-92	Pass
*1-FW-S5	08-22-92	08-27-92	Pass
1-FW-S6	08-13-92	09-02-92	Pass
1-MSS-7	08-19-92	08-26-92	Pass
1-MSS-8	08-19-92	09-08-92	Pass
1-GCCW-S844	07-31-92	08-05-92	Pass
1-GSI-S128	08-06-92	08-14-92	Pass
1-GCTS-S160A(N)	08-14-92	08-20-92	Pass
1-GCTS-S161(E)	08-07-92	08-14-92	Pass
1-GCTS-S161(W)	08-07-92	08-14-92	Pass
S/N 25.12620.007-1	07-18-92	09-12-92	Pass
S/N 25.12620.007-5	07-18-92	09-12-92	Pass
S/N 25.12620.007-6	07-18-92	09-12-92	Pass

* These snubbers were tested due to functional failures at the previous testing period (last outage).

AMERICAN ELECTRIC POWER COMPANY
 C. COOK NUCLEAR PLANT, BRIDGMAN, MI
 UNIT 1 COMMERCIAL SERVICE DATE 8-23-1977
 NATIONAL BOARD NUMBER 20761

UNIT 1 GRINNELL "INACCESSIBLE" SNUBBERS

Tech. Spec. Mark Number Azimuth Elevation Size(In.)

1	1-GRC-S519	310	683	2 1/2 X 5
2	1-GRC-S537	25	610	2 1/2 X 5
3	1-GRC-S538	41	614	2 1/2 X 5
4	1-GRC-S555	141	614	2 1/2 X 5
5	1-GRC-S562	154	610	2 1/2 X 5
6	1-GRC-S564	313	614	2 1/2 X 5
7	1-GRC-S566	332	610	2 1/2 X 5
8	1-GRC-S573	223	614	2 1/2 X 5
9	1-GRC-S575	208	610	2 1/2 X 5
10	1-GRC-S582	212	617	1 1/2 X 5
11	1-GRC-S587	260	622	2 1/2 X 5
12	1-GRC-S592	292	683	3 1/4 X 5
13	1-GRC-S594	292	691	3 1/4 X 5
14	1-GRC-S596	285	691	2 1/2 X 5
15	1-GRC-S598	292	670	2 1/2 X 5
16	1-GRC-S599	287	672	2 1/2 X 5
17	1-GRC-S604	286	688	1 1/2 X 5
18	1-GRC-S608	286	693	1 1/2 X 5
19	1-GRC-S614	681	681	1 1/2 X 5
20	1-FW-S1	31	634	2 1/2 X 5
21	1-FW-S2(L)	26	633	2 1/2 X 5
22	1-FW-S2(U)	26	636	2 1/2 X 5
23	1-FW-S3	20	629	2 1/2 X 5
24	1-FW-S4(L)	154	636	2 1/2 X 5
25	1-FW-S4(U)	154	640	2 1/2 X 5
26	1-FW-S5	163	634	2 1/2 X 5
27	1-FW-S6	157	629	2 1/2 X 5
28	1-FW-S7	204	634	2 1/2 X 5
29	1-FW-S8(L)	200	633	2 1/2 X 5
30	1-FW-S8(U)	200	636	2 1/2 X 5
31	1-FW-S9	194	629	2 1/2 X 5
32	1-FW-S10(L)	334	633	2 1/2 X 5
33	1-FW-S10(U)	334	636	2 1/2 X 5
34	1-FW-S11	330	634	2 1/2 X 5
35	1-FW-S12	343	629	2 1/2 X 5
36	1-GCS-S634	292	613	1 1/2 X 5
38	1-GCS-S757	128	612	1 1/2 X 5
39	1-MSS-1	8	639	5 X 5
40	1-MSS-2	17	635	3 1/4 X 5
41	1-MSS-3	172	639	5 X 5
42	1-MSS-4	165	635	3 1/4 X 10
43	1-MSS-5	191	635	3 1/4 X 5

DOW CHEMICAL COMPANY
 C. COOK NUCLEAR PLANT, BRIDGMAN, MI
 UNIT 1 COMMERCIAL SERVICE DATE 8-23-1977
 NATIONAL BOARD NUMBER 20761

44	1-MSS-6	184	639	4 X 5
45	1-MSS-7	349	635	3 1/4 X 5
46	1-MSS-8	356	639	4 X 5
70	1-QSI-S707	228	608	1 1/2 X 5

INDIANA MICHIGAN POWER COMPANY
 J.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
 UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
 NATIONAL BOARD NUMBER 20761

UNIT 1 GRINNELL "ACCESSIBLE" SNUBBERS

<u>Tech. Spec.</u>	<u>Mark Number</u>	<u>Azimuth</u>	<u>Elevation</u>	<u>Size(in.)</u>
37	1-GCS-S637	71	608	1 1/2 X 10
47	1-GCCW-S278		609	1 1/2 X 5
48	1-GCCW-S309		597	4 X 5
49	1-GCCW-S837		621	1 1/2 X 5
50	1-GCCW-S838		621	1 1/2 X 5
51	1-GCCW-S839		621	1 1/2 X 5
52	1-GCCW-S840		621	1 1/2 X 5
53	1-GCCW-S841		621	1 1/2 X 5
54	1-GCCW-S842		621	1 1/2 X 5
55	1-GCCW-S844		609	1 1/2 X 5
56	1-GBD-S563	277	608	1 1/2 X 5
57	1-GBD-S569	278	608	1 1/2 X 5
58	1-GBD-S573	181	607	1 1/2 X 5
59	1-GBD-S574	181	607	1 1/2 X 5
60	1-GRH-S7A		581	2 1/2 X 5
61	1-GRH-S7B		581	2 1/2 X 5
62	1-GRH-S47		581	2 1/2 X 5
63	1-GRH-S48		580	2 1/2 X 5
64	1-GDG-S13		596	2 1/2 X 5
65	1-GDG-S14		603	2 1/2 X 5
66	1-GSI-S103		573	3 1/4 X 5
67	1-GSI-S128		573	3 1/4 X 5
68	1-GSI-S575	66	598	2 1/2 X 5
69	1-GSI-S657	185	610	1 1/2 X 5
71	1-GCTS-S73 (E)		582	2 1/2 X 5
72	1-GCTS-S73 (W)		582	2 1/2 X 5
73	1-GCTS-S76 (B)		579	2 1/2 X 5
74	1-GCTS-S76 (A)		579	2 1/2 X 5
75	1-GCTS-S160A(N)		582	2 1/2 X 5
76	1-GCTS-S160A(S)		582	2 1/2 X 5
77	1-GCTS-S161 (E)		579	2 1/2 X 5
78	1-GCTS-S161 (W)		579	2 1/2 X 5

ISI LONG TERM PLAN SNUBBERS

1-GFW-S852
1-GFW-S854
1-GFW-S861
1-GFW-S862
1-GFW-S866

INDIANA MICHIGAN POWER COMPANY
J.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

INDIANA MICHIGAN POWER COMPANY
COOK NUCLEAR PLANT, BRIDGMAN, MICHIGAN
UNIT ONE COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

STEAM GENERATOR INSPECTIONS AND REPAIRS

Unit One Steam Generator inspections and repairs were performed from July 11, 1992 to September 11, 1992 in accordance with plant Technical Specifications. Eddy current bobbin coil testing was performed on 100% of the inservice tubes in all four steam generators. Selected areas of the tubes were also inspected with rotating pancake (RPC) probes. All tubes with indications exceeding Tech Spec limits were repaired either by sleeving or plugging. Listed below are the number of tubes repaired per steam generator:

Steam Generator 11:

827 sleeves installed in the hot leg
135 tubes removed from service by plugging

Steam Generator 12:

180 sleeves installed in the hot leg
96 tubes removed from service by plugging

Steam Generator 13:

459 sleeves installed in the hot leg
72 tubes removed from service by plugging

Steam Generator 14:

374 sleeves installed in the hot leg
51 tubes removed from service by plugging

INDIANA MICHIGAN POWER COMPANY
J.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

SYSTEM PRESSURE TESTS

Inservice Inspection System Pressure Tests were performed during the Summer 1992 Unit One Refueling Outage. These tests were in accordance with the requirements of ASME Section XI Articles IWA-5000, IWB-5000, IWC-5000, and IWD-5000 as applicable.

A total of thirty-eight (38) hydrostatic tests were performed during the outage with the results of all tests within allowable acceptance criteria. The tests performed were:

<u>PROCEDURE</u>	<u>DESCRIPTION</u>	<u>NUMBER OF TESTS COMPLETED</u>
**1-OHP 5070.ISI.006	Pressurizer and Reactor Head Post Accident Vents	2
**1-OHP 5070.ISI.009	Component Cooling Water; East and West Trains	2
**1-OHP 5070.ISI.010	Component Cooling Water Piping Misc Supply/Return Headers	1
**1-OHP 5070.ISI.011	Component Cooling Water Piping at CPN-25 and 72 serving CPN & CEQ Fan Motor Cooling	4
**1-OHP 5070.ISI.012	Component Cooling Water Piping Excess Letdown Heat Exchanger, RCP Thermal Barriers, Oil Coolers	3
**1-OHP 5070.ISI.013	Non-Essential Service Water: CPN's to/from Containment Ventilation Units	13
**1-OHP 5070.ISI.020	RHR: Normal Pump Suction	1

INDIANA MICHIGAN POWER COMPANY
J.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

<u>PROCEDURE</u>	<u>DESCRIPTION</u>	<u>NUMBER OF TESTS COMPLETED</u>
**1-OHP 5070.ISI.023	Nitrogen Piping to Accumulators at CPN-32	1
**1-OHP 5070.ISI.045	Containment Purge Piping at CPN's 59,61,62,63,64,65	7
**1-OHP 5070.ISI.048	Plant and Control Air Piping at CPN's	4

In addition system functional, inservice and leakage tests were conducted on class one, two and three systems. This included the entire Reactor Coolant System and associated piping per the requirement of IWB-5221. The results of the system functional, inservice and leakage tests were all within allowable acceptable range.

All procedures satisfy the testing and documentation requirements set forth in ASME Section XI, 1983, Summer 1983 Addendum for Class 1, Class 2, and Class 3 components and systems for Cook Nuclear Plant.

All procedures were satisfactorily completed by Indiana Michigan Power Company, ISI Hydro Testing, Quality Control, and NDE Personnel.

INDIANA MICHIGAN POWER COMPANY
J.C. COOK NUCLEAR PLANT, BRIDGMAN, MI
UNIT 1 COMMERCIAL SERVICE DATE 8-23-1975
NATIONAL BOARD NUMBER 20761

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RECIP. NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards "Inservice Insp Rept for Cycle 8/9 Refueling
Outage," as required by Article IWA-6000 of Section XI of
ASME Boiler & Pressure Vessel Code.

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*Not a code designated report,
+ this is plan. Copies of code figures
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*Dynacon
prepared*

33 pages

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Ralph E. Beedle
Executive Vice President
Nuclear Generation

October 26, 1992
IPN-92-052

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Inservice Inspection Report for
Cycle 8/9 Refueling Outage

Dear Sir:

This letter submits the Inservice Inspection (ISI) Report (Attachment I) as required by Article IWA-6000 of Section XI of the ASME Boiler and Pressure Vessel Code.

This inspection is the second examination of the second 40-month period of the Second 10-Year Indian Point 3 ISI Program Interval and was conducted during the Cycle 8/9 Refueling Outage. The report consists of extracts from Tabs A, B, and C of the Final Weld and Component Examination ISI Report.

If you have any questions, please contact Mr. P. Kokolakis.

wrong extracts
Very truly yours,

[Signature]
Ralph E. Beedle

Attachment.

cc: see next page

200019

9211030336 921026
PDR ADOCK 05000286
Q PDR

C-180

AC47

NEW YORK POWER AUTHORITY
INDIAN POINT UNIT NO. 3
2ND OUTAGE 2ND PERIOD 2ND INTERVAL
1992

INTRODUCTION

An Inservice Examination Program was performed at the Indian Point Unit No. 3 Nuclear Power Plant from August 23, 1991 thru May 30, 1992 by Westinghouse Electric Corporation, Dynacon Systems Inc. and New York Power Authority - Indian Point Unit No. 3, of Class 1, Class 2 and Class 3 components and piping. The examinations were performed in accordance with an approved Examination Program Plan located under Tab C of the Final Report.

Examinations were performed to satisfy the requirements of:

- o ASME Boiler and Pressure Vessel Code Section XI, 1983 Edition up to and including Summer 1983 Addenda
- o ASME Boiler and Pressure Vessel Code Section XI, 1986 Edition, Appendix VI
- o United States Nuclear Regulatory Commission, Nuclear Regulatory Guide 1.14
- o Surface examination of Class 1 Steam Generator Manway Bolting per Indian Point Unit No. 3 commitment to the United States Nuclear Regulatory Commission

Examination procedures were approved prior to the examinations, and certification documents relative to personnel, equipment and materials were reviewed and determined to be satisfactory. Inspection, review, witnessing and surveillance of the examinations and related activities were conducted by personnel from: The Nuclear Regulatory Commission, the Hartford Steam Boiler Inspection and Insurance Company and the New York Power Authority Quality Assurance Department.

RESULTS

Examinations based on the Examination Program Plan noted 36 recordable indication areas on the basis of Westinghouse procedure recording criteria, which, generally are more critical than specified Section XI acceptance criteria. These indications consisted of (a) 10 surface, which were either accepted or eliminated during investigation and (b) 26 visual of which all were IWF components. All indications were evaluated and corrected or accepted as appropriate to maintain integrity of the individual components.

Specific data relative to all the indications including dispositions, NCR's and Acceptance Standards are located under Tab D and Tab F of the Final Report.

EXAMINATION

Examinations were conducted to inspect as much of the specified surface/volume as reasonably practical. In some instances, examinations were limited by geometric, metallurgical or design access restrictions. In each case, the occurrence and the cause of various limitations have been noted and the resulting coverage is the maximum that is reasonably achievable.

Some of the arrangements and details of the piping systems and components were designed and fabricated before the access and examination requirements of Section XI Code were established. Specified limitations and restrictions for all examinations are indicated on the examiners data sheets, under Tab D of the Final Report.

The following items were examined by New York Power Authority Indian Point No. 3 personnel:

- o Class 1 System Leakage Test
- o Class 2 System Leakage Test
- o Class 3 System Leakage Test

James R. Delbusso 6-30-92
James R. Delbusso Date
Westinghouse Nuclear Service Division
Dynacon Systems Inc. ISI Coordinator

Inservice Examination Program Plan
For
New York Power Authority
Indian Point Unit No.3
2nd Outage: 2nd Period: 2nd Interval
1992

This document details the proposed planned scope of examination by Westinghouse Nuclear Services Division Inspection Services for the 2nd Outage, 2nd Period, 2nd Interval, including items and areas selected for examination and documentation procedures and sketches containing identifications of all areas to be examined.

Qualifications of examiners, materials and equipment will be available on site prior to the start of examinations.

Efforts should be made to provide access to all planned examinations, however, due to circumstances such as environment, access, radiation levels, etc., some variations may occur.

This program and the procedures incorporated herein require approval of New York Power Authority and the Authorized Inspector prior to the start of examinations.

Prepared By: WESTINGHOUSE ELECTRIC CORPORATION

W Approval: James R. Dellurra 4-23-92
Site Coordinator Date

New York Power
Authority Approval: [Signature] 4/23/92
Date

Title: SITE NDE LEVEL III

Authorized Inspector
Review: [Signature] 4/23/92
Date

Agency: A.S.B. I & I Co.

*plan only,
no results*

**NEW YORK POWER AUTHORITY
INDIAN POINT UNIT NO. 3
2ND OUTAGE; 2ND PERIOD; 2ND INTERVAL
INSERVICE EXAMINATION PROGRAM
1992**

All items listed below were examined, as indicated, in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section XI-1983 Edition up to and including Summer 1983 Addenda and Plant Technical Specification to the extent practical with the access provided and the limitations of component geometry.

Program Item	IWB-2500-1 Reference	Area and Extent of Examination	Examination Procedure			Sketch Reference
			Vol.	Surf.	Vis	
<u>Reactor Vessel</u>						
1.	B6.30	Closure Head Studs - 19 thru 36	115	--	--	1-1400
2.	B6.30	Closure Head Stud-Replacement 18	115	70	--	1-1400
3.	B15.10	Pressure Retaining Boundary	--	--	(1)	(1)
<u>Pressurizer</u>						
4.	B2.11	Circumferential Weld 17	--	--	(2)	(2)
5.	B2.12	Longitudinal Weld 16	--	--	(2)	(2)
6.	B3.120	Nozzle Inside Radius Section - 20IR, 21IR, 22IR, 23IR, 24IR and 25IR	--	--	(2)	(2)
7.	B15.20	Pressure Retaining Boundary	--	--	(1)	(1)
<u>Replacement Steam Generator 31, 32, 33 and 34</u>						
8.	B7.30	Manway Bolting-4 From Each Hotleg and Coldleg Manway	--	70	--	1-3101 (4)
9.	B15.30	Pressure Retaining Boundary	--	--	(1)	(1)
<u>Pressure Retaining Bolting</u>						
10.	B7.50	6" Pressurizer Safety-Flange A- 12 Studs and 24 Nuts	--	--	8	1-4503
<u>Piping</u>						
11.	B15.50	Pressure Retaining Boundary	--	--	(1)	(1)

Public Service
Electric and Gas
Company

Stanley LaBruna

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609-339-1200

Vice President - Nuclear Operations

NOV 16 1992

NLR-N92165

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

INSERVICE INSPECTION ACTIVITIES - 90 DAY REPORT
TENTH REFUELING OUTAGE
SALEM GENERATING STATION
UNIT NO. 1
DOCKET NO. 50-272

Public Service Electric & Gas Company (PSE&G) hereby provides in the enclosure to this letter, the Ninety (90) Day Report for Inservice Inspection (ISI) activities conducted at Salem Generating Station Unit No. 1 during the tenth refueling outage. This report is submitted in accordance with Section 4.0.5 of Appendix A to the Technical Specifications for Salem Generating Station, Unit No. 1 and Article IWA-6220(b) of Section XI of the ASME Boiler and Pressure Vessel Code.

Enclosure 1 of this submittal consists of Form NIS-1, "Owners Data Report for Inservice Inspection", Forms NIS-2 and/or NR-1 "Owner's Report for Repairs or Replacements," and pertinent sections of Volume 1 of the "Examination of Selected Components at Salem Generating Station Unit No. 1".

Should there be any questions with regard to this submittal, please do not hesitate to contact us.

Sincerely,

Enclosure

230050

C-185

9211240264 921116
PDR ADOCK 05000272
Q PDR

AC47

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit: N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
8. This report covers the examinations that were conducted
during the 10th Refueling Outage.
9. Components Inspected.

COMPONENTS OR APPURTENANCE	MANUFACTURER OR INSTALLER	MANUFACTURER OR INSTALLER SER. NO.	STATION OR PROVINCE NUMBER	NATIONAL BOARD NO.
#1 Reactor Vessel	Combustion Engineering	66201 Head 66101 Vessel	N/A	20757
#11 Steam Generator	Westinghouse Tampa Div. P.O. Box 19218 Tampa, FL 33616	1003	N/A	68-10
#12 Steam Generator	" "	1022	N/A	68-08
#13 Steam Generator	" "	1023	N/A	68-09
#14 Steam Generator	" "	1203	N/A	68-51
#11 Regen. Ht. Exch.	Sentry Equip. Corp. Oconomowoc, WI	4195-A4 7, 8, 9	N/A	385, 386, 387

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
9. Components Inspected (cont'd):

COMPONENTS OR APPURTENANCE	MANUFACTURER OR INSTALLER	MANUFACTURER OR INSTALLER SER. NO.	STATION OR PROVINCE NUMBER	NATIONAL BOARD NO.
#1 Pressurizer	Delta Southern	1011	N/A	68-8
Chemical Volume & Control Piping System	United Engineers & Constructors (UE&C) 30 South 17th St., Phila. PA 19101	N/A	N/A	N/A
Containment Spray Piping System	UE&C	N/A	N/A	N/A
Mainsteam Piping sys.	UE&C	N/A	N/A	N/A
#1 Volume Control Tank	Joseph Oat & Sons, Inc. Camden, N.J.	1781-2A	N/A	374
#1 Excess Letdown Heat Exchanger	Atlas Industries 81 Somerset Place Clifton, N.J.	850	N/A	701
#1 Letdown Heat Exchanger	Atlas Industries	N/A	N/A	694
#1 RHR Heat Exchanger	Engineers & Fabricators, Inc.	S15860-C	N/A	122

*all this on
every page is a
waste of space and
produces an unneeded
paperwork burden.*

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
9. Components Inspected (cont'd).:

COMPONENTS OR APPURTENANCE	MANUFACTURER OR INSTALLER	MANUFACTURER OR INSTALLER SER. NO.	STATION OR PROVINCE NUMBER	NATIONAL BOARD NO.
Pressurizer Relief Piping Sys.	UE&C	N/A	N/A	N/A
Reactor Coolant Piping Sys	UE&C	N/A	N/A	N/A
Steam Gen. Feed Piping System	UE&C	N/A	N/A	N/A
Safety Injection Piping Sys.	UE&C	N/A	N/A	N/A

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
10. Abstracts of Examinations:

- 10.1 Examinations were conducted by PSE&G as well as companies under contract to PSE&G. The following examination summary (Section 11) is a brief summary with further details found in the attached report and on file at the Salem Generating Station.

Since nearly all inservice examination requirements can be found in Salem's Technical Specifications, this summary lists these requirements first arranged by Tech. Spec. paragraph number followed by applicable NRC circulars and bulletins.

11. Examination Summary:

11.1 Technical Specification 4.0.5 - ASME Section XI

The examinations were conducted during the tenth refueling ISI outage at Salem Unit 1. These examinations constituted the third ISI performed at Salem Unit 1 during the second - ten year interval of operation. The components were examined in accordance with "Examination Plan for the 1992 Inservice Examination of Salem generating Station, Unit 1".

**ASME CODE BOOK 3 DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.1 (cont.)

11.1.1 ISI Examination

Southwest Research Institute (SWRI) under the direction of the Salem ISI Group conducted eighty-two (82) ultrasonic, one hundred thirty-one (131) liquid penetrant, fifty-six (56) magnetic particle, and one (1) visual examination. The examinations were conducted on the following welds and components:

Class 1 Components

Vessels

Reactor Pressure Vessel Closure head
Pressurizer

Piping

Chemical and Volume Control System
Pressure Relief System
Pressurizing System
Reactor Coolant System
Residual Heat Removal System
Safety Injection System

Pumps

Reactor Coolant Pump Bolts and Flywheels

Class 2 Components

Vessels

Steam Generator
Boron Injection Tank

Piping

Feedwater System
Main Steam System

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):
 - 11.1.2 UT examination of #12 Steam Generator transi-
tion cone to shell weld per PSE&G's response to
I.E. Notice 90-04.
 - 11.1.3 The following recordable indications were
reported during the course of the examinations.
 - A. During the PT examinations, recordable
indications were revealed in nine examina-
tion areas. These indications were report-
ed to PSE&G on Customer notification Forms
(CNFs) SAM 1-1 through 1-6, 1-6A, 1-6B, 1-
6C and 1-7 through 1-9. One indication was
resolved as "Code allowable" and accepted
"as is". Seven indications were removed
with cosmetic buffing. One was mechanical-
ly removed after confirmation and accept-
ance with the UT method.
 - B. During the MT examinations, recordable
indications were revealed in three areas.
One was resolved as "code allowable" and
accepted "as is". One indication was
mechanically removed after sizing with the
UT method. The other was recorded as being
outside the American Society of Mechanical
Engineers Section XI examination area on
the adjacent valve casting. These indica-
tions were reported to PSE&G on CNFs SAM 1-
10, 1-11, 1-11A, 1-11B and 1-12.
 - 11.1.4 There were no other reportable indications
identified during this ISI.

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.1.5 Visual Examinations

- A. Contract personnel under the direction of the Salem ISI Group conducted visual examinations on seventy-six Nuclear Class I, eighty-two Nuclear Class II eighty-eight Nuclear Class III supports. There were several minor discrepancies noted including; wrong spring settings, rust, loose bolts, etc.. Work orders and/or deficiency reports were generated to document and correct all discrepancies. As a result of some of the discrepancies, twenty-one additional support exams were scheduled and performed satisfactory.
- B. PSE&G conducted VT-1 visual examinations of Nuclear Class 1 bolting on five components (valves and/or flanges) satisfying ASME Section XI B-G-2 examination requirements. Results of these examinations were satisfactory.
- C. PSE&G conducted VT-3 visual examinations of five components (vessel and/or pump) supports to satisfy examination category D-A, D-B, and F-A of ASME Section XI. Results of these examinations were satisfactory.

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.1.6 System Pressure Tests

Summary

System leakage exams were performed in accordance with ASME Section XI on all Nuclear Class 1 systems.

System functional Tests/system inservice tests were performed on those Nuclear Class 2 and 3 systems required to satisfy a percentage of the requirements for the second inspection period.

Results

None of the systems tested exhibited any excessive leakage. There was some minor mechanical leakage which was identified and work orders were written to correct.

11.2 Technical Specification 4.7.9 (a) Visual Inspection of Hydraulic and Mechanical Snubbers

11.2.1 Summary

Contract and PSE&G personnel under the direction of the Salem ISI Group conducted visual examinations on one hundred and sixteen mechanical snubbers and twenty-four hydraulic snubbers.

11.2.2 Results

There were no discrepancies noted which would affect snubber operability. There were some minor items noted including reservoirs not full, missing washers, loose spherical bearings and binding around the bearings, and some light corrosion. These conditions were all noted and corrected.

All snubbers which were removed and reinstalled for functional testing received an "As-Left" visual examination.

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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Newark, N.J. 07101
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Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

**11.3 Technical Specification 4.7.9 (c) "Functional Testing
of Selected Hydraulic and Mechanical Snubbers"**

11.3.1 Summary - Hydraulic Snubbers

Per Technical Specifications, 10% of the installed hydraulic snubbers as well as all previous outage functional test failures are required to be tested. The 10% sample included one 200 KIP Main Steam snubber and two 1000 KIP Steam Generator snubbers. There were no previous outage test failures.

11.3.2 Results - Hydraulic Snubbers

There were noted discrepancies in the Tech. Spec. sample.

11.3.3 Summary - Mechanical Snubbers

Per Technical Specifications, 10% of the total installed mechanical snubbers as well as previous outage functional test failures were tested utilizing the WYLE Model 150 Snubber Testing machine. The 10% sample is broken down by snubber category size (small and medium) with 10% from each size selected each outage. All selected snubbers are detailed below:

- A. The 10% sample of the medium size snubbers included four (4) PSA-1's, six (6) PSA-3's, one (1) PSA-10 and one (1) PSA-35.
- B. Due to previous outage failures, 100% of the small size snubbers (PSA-1/4's, PSA-1/2") and equiv. hydraulic snubbers are tested each outage. This resulted in tests being performed on one Model 1/4, three Model 1/2 and one Lisega HSML.

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.3.4 Results - Mechanical Snubbers

- A. All the medium size Tech. Spec. sample
snubbers were tested satisfactory.
- B. All the small size snubber and equivalent
hydraulic snubbers were tested satisfactory.

**11.4 NRC Circular 76-06 and NRC Bulletin 79-17 "Commitment
to Examine and/or Flush Stainless Steel Lines Contain-
ing Borated Water.**

**11.4.1 Southwest Research Institute (SwRI) under the
direction of the Salem ISI Group performed
ultrasonic examinations on fourteen welds. No
recordable indications were noted.**

**11.4.2 Quarterly chemistry samples were taken by the
PSE&G Chemistry Department and the results were
transmitted to the Salem ISI Group per the
Chemistry procedure. During the past 18
months, this program identified a few noncon-
formances. The nonconforming pipe was flushed
and a new sample taken to verify it met the
acceptance criteria.**

**11.5 Technical Specification 4.4.11.1 Reactor Coolant Pump
Flywheel Examination (USNRC Regulatory Guide 1.14,
"Reactor Coolant Pump Flywheel Integrity").**

11.5.1 Summary

Southwest Research Institute (SwRI) performed
examinations on Reactor Coolant Pump (RCP)
Flywheels. Flywheels are tracked by RCP Motor
serial number and original pump location,
however, many have been moved to different pump
locations throughout outages. The exam results
are as follows:

- A. Performed "UT" examination of the top side
of serial # 35-76P-917 which was originally
installed in #23 pump and is now installed
in #12 pump. Exam performed in-place.

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

B. Performed "UT" examination of the top side of serial # 45-75P-790 which was originally installed in #14 pump and is now installed in #11 pump. Exam performed in-place.

11.5.2 Results

No discrepancies were noted.

11.6 NRC Bulletin 87-01 "Thinning of Pipe Walls in Nuclear Power Plants".

11.6.1 Summary

Southwest Research Institute (SwRI) and General Electric Co. under the direction of the Salem ISI Group performed ultrasonic thickness measurements on eighty-one piping components as identified by PSE&G engineering.

11.6.2 Results

Any areas identified having a wall thickness below minimum wall were submitted to PSE&G Engineering Department for disposition.

11.7 NRC Bulletin 82-02 "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of Power Plants".

11.7.1 Summary

Visual (VT) and Magnetic Particle (MT) exams were performed on the primary manway bolts for all four Steam Generators. In addition, a VT exam was performed on all four Reactor Coolant Pumps main flange bolting.

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.7.2 Results

One bolt on #14 Steam Generator hot leg had minor indications observed during the visual examinations. Deficiency Reports and Code Job Packages were prepared to replace these bolts in kind. There were no discrepancies noted on the Reactor Coolant Pump flange bolting.

11.8 NRC Bulletin 88-08 "Thermal Stresses in Piping Connected to reactor Coolant System".

11.8.1 Southwest Research Institute (SwRI), under the direction of the Salem ISI Group, performed augmented ultrasonic and dye penetrant examinations on seventeen 1.5" Safety Injection welds, two 2" and two 3" Chemical and Volume Control System welds for the third consecutive outage. No discrepancies were noted.

11.9 NRC Bulletin 88-11 "Pressurizer Surge Line Thermal Stratification".

11.9.1 Summary

The Salem ISI Group performed visual examinations on all the existing surge line pipe supports\HEBA restraints and any insulation damage. A total of nine supports were examined.

11.9.2 Results

No discrepancies were noted. Results were forwarded to system engineering.

**FORM NDT-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

**11.10 Technical Specification 4.4.5.0 - Steam Generator Tube
eddy current inspection.**

**11.10.1 The original eddy current examination scope is
detailed below:**

- A. Three percent Tech. Spec bobbin coil examination in #11, #12, #13 and #14 steam generators (102 tubes per steam generator).
- B. A non-tech spec sample of six hundred ten tubes in #11 steam generator per Engineering's recommendation to comply with EPRI recommended guidelines of performing 100% examination of all steam generator tubes over a six year period. This sample combined with the tech Spec sample brings the total percentage of tubes inspected in this generator for this outage to 21%.
- C. A non-tech. spec. sample of six hundred forty-one tubes in #12 steam generator per Engineering's recommendation to comply with EPRI recommended guidelines of performing 100% examination of all steam generator tubes over a six year period. This sample combined with the tech. Spec. sample brings the total percentage of tubes inspected in this generator for this outage to 22%.
- D. A non-tech. spec sample of six hundred and nine tubes in #13 steam generator per Engineering's recommendation to comply with EPRI recommended guidelines of performing 100% examination of all steam generator tubes over a six year period. This sample combined with the Tech. Spec. sample brings the total percentage of tubes inspected in this generator for this outage to 21%.

FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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2. Plant: Salem Generating Station P.O. Box 236,
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3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

- D. A non-tech. spec sample of six hundred thirty-one tubes in #14 steam generator per Engineering's recommendation to comply with EPRI recommended guidelines of performing 100% examination of all steam generator tubes over a six year period. This sample combined with the Tech. Spec. sample brings the total percentage of tubes inspected in this generator for this outage to 21%.
- E. A twenty percent inspection (approximately 680 tubes) in #11, #12, #13 and #14 steam generators with rotating pancake probe to address the concern of Expansion Zone Pure Water Stress Corrosion Cracking at the hot leg tube sheet (WETEX) for steam generators manufactured using explosive tube expansion techniques.
- F. A seven percent inspection (approximately 240) in #11, #12, #13 and #14 steam generators with rotating pancake probe at multiple support plate intersections.

11.10.2 The results of the eddy current inspections are as follows:

- A. The Tech. Spec. sample in #11, #12, #13 and #14 steam generators revealed no indication equal to or exceeding the plugging limit and satisfied the less than 5% of the inspected tubes being degraded therefore no scope expansion was required.
- B. Zero tubes in #11 steam generator were required to be plugged due to eddy current indications.
- C. One tube in #12 steam generator was required to be plugged due to an eddy current indication on the non-tech. spec. sample exceeding the plugging limit.

FORM NDT-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

D. Two tubes in #13 steam generator were required to be plugged due to eddy current indications in the non-tech. spec. sample exceeding the plugging limit. In addition, one tube was preventively plugged due to an eddy current indication in the 35% to 39% through wall range.

E. Zero tubes in #14 steam generator were required to be plugged due to eddy current indications exceeding the plugging limit however two tubes were preventively plugged due to eddy current indications in the 35% to 39% through wall range.

**ASME REPORT ON THE SERVICE INSPECTION
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

1. Owner: Public Service Electric & Gas Co., 80 Park Plaza
Newark, N.J. 07101
2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

Steam Generator	Extent of Exam.	No. of Tubes Inspected	Tubes with Indications		
			< 20%	20-39%	> 40%
11	Full Length Bobbin Coil	712	9	16	0
	RCP at Support Plates	237	0	0	0
	RCP of Tubesheet	680	0	0	0
12	Full Length Bobbin Coil	743	20	41	1
	RCP at Support Plates	240	0	0	0
	RCP of Tubesheet	678	0	0	0
13	Full Length Bobbin Coil	711	8	10	1
	RCP at Support Plates	241	0	0	0
	RCP of Tubesheet	687	0	0	0
14	Full Length Bobbin Coil	733	15	39	0
	RCP at Support Plates	241	0	0	0
	RCP of Tubesheet	678	0	0	0
Sub-Total	Full Length	2899	52	106	2
	Support Plates	959	0	0	0
	Tubesheet	2723	0	0	1
TOTAL		6581	52	106	3

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

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2. Plant: Salem Generating Station P.O. Box 236,
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3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.10.3 Additional Steam Generator Work

- A. Helium leak detection was performed on #11 steam generator to locate the source of primary to secondary leakage. No positive evidence of primary to secondary leakage was observed using the Helium Leak Detection Technique however tubesheet video scan during helium pressurization revealed an excessive amount of boron deposits observed surrounding the explosive plugs located in Row 1 Column 91 hot leg and Row 1 Column 2 in the cold leg. The explosive plugs in these locations were remotely machined and replaced with short standard type Inconel 690 mechanical plugs.
- B. All explosive plugs in both legs of #11 steam generator were scheduled for replacement. These plugs were located at Row 1 Columns 1-5 and 90-94 in both legs, twenty total. Due to difficulties encountered while attempting to machine the explosive plugs and time constraints, only six of the twenty plugs were removed. Four of these plugs were reported as having both boron deposits surrounding them during helium pressurization and were successfully removed (see above). In addition the explosive plugs in Row 1 Column 5 hot and cold were removed and replaced with a short standard type Inconel mechanical plug.

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

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2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.10.4 Current status of Steam Generators

At this time, the total number of tubes plugged
in the Salem Unit 1 Steam Generators are as
follows:

	S/G 11	S/G 12	S/G 13	S/G 14
Plugs repaired this outage	6	0	0	0
New tubes plunged this outage	0	1	3	2
Tubes previously plugged	104	120	109	141
Total tubes plugged	104	121	112	143

**11.11 Technical Specification 4.6.1.2 (d) Containment Type
"B" (Penetrations) and "C" (Valves) Leak Rate Testing.**

11.11.1 Summary

The PSE&G Research Corp. under the field supervision of the Salem ISI Group, conducted leak rate tests on sixty-three Type "B" Electrical Penetrations, seven Type "B" Mechanical Penetrations and fifty-five Type "C" Mechanical penetrations (113 valves).

11.11.2 Results

- A. Several Type "C" valves exceeded the recommended leak rate acceptance criteria. These valves were reworked to an acceptable condition and retested satisfactorily.
- B. A total As-Found leakage of 32,432.7 SCCM was measured through all Type "B" and "C" containment penetrations. After corrective maintenance, a total As-Left leakage of 22,338.2 was measured through all Type "B" and "C" penetrations.

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
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3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
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6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

**11.12 Technical Specification 4.6.1.3b Elevation 100' and
130' Airlock Leak Rate Tests**

11.12.1 Summary

PSE&G Research Corp. under the field supervision of the Salem ISI Group, conducted leak rate tests on the 100' and 130' containment personnel airlocks.

11.12.2 Results

The combined As-Found leakage for these airlocks was 8,340.0 SCCM. After scheduled preventative maintenance, the combined As-Left leakage for the airlocks was 9,500.0 SCCM. Both of these measured combined leakage rates satisfy the Technical Specification acceptance criteria of ≤ 0.05 La (10,812 SCCM).

- 11.13 NUREG 0578 - "TMI Lessons Learned - Performing Service Leak Exams, measuring leakage outside Containment, and take corrective actions necessary to reduce leakage as low as possible on systems likely to contain radioactive liquids in the event of an accident.**

11.13.1 Summary

- A. The Salem ISI Group performed service pressure leak exams on the Safety injection, Chemical and Volume Control, Residual Heat Removal and Containment Spray Systems.
- B. The PSE&G Research Corp., under the direction of the Salem ISI Group conducted the Waste Gas System Leak Test in accordance with Site Service procedure SC.SS-IS.WG-0003(R).

**FORM NIS-1 OWNER'S DATA REPORT FOR INSERVICE INSPECTIONS
AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES**

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2. Plant: Salem Generating Station P.O. Box 236,
Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

11.13.2 Results

- A. The Waste Gas Decay Tanks, Waste Gas Suction and Discharge Headers and #11 Waste Gas Compressor showed no adverse findings relative to this bulletin.
- B. #12 Waste Gas Compressor could not be tested because it is out of service awaiting breaker maintenance. The activity to perform a leak rate check remains open until the breaker is repaired.
- C. The leak rate test on the 1WG41 valve was acceptable with a measured leakage of 1,629 SCCM which is below the acceptance criteria of 1,850 SCCM.

11.14 Field Directive, S-C-R700-MFD-288, "Leak Rate test Program for the Steam Generator Blowdown Isolation Valve (GB4)

11.14.1 PSE&G Research and Testing Lab under the field supervision of the Salem ISI Group, conducted leak rate tests on all four Steam Generator Blowdown Valves (GB-4's) in accordance with Site Service procedure SC.SS-IS.GB-0004(Q).

11.14.2 Results

All four valves had an acceptable leak rates below the test acceptance criteria of $\leq 30,000$ SCCM. No discrepancies were noted.

AS REQUIRED BY THE PROVISIONS OF THE ASME CODE RULES

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Hancock's Bridge, N.J. 08038
3. Plant Unit 1
4. Owner's Certificate of Authorization (if required) N/A.
5. Commercial Service Date 7/1/77.
6. National Board Number for Unit N/A.
7. Examination Dates: 2-26-91 through 4-17-92.
11. Examination Summary (cont'd):

We certify that the statements made in this report are correct and the examinations and corrective measures taken conform to the rules of the ASME Code, Section XI.

Date Nov 3, 19 92 Signed PSE&G Owner By John F. Harrison

Certificate of Authorization No. (if applicable) N/A

Expiration Date N/A

CERTIFICATE OF INSERVICE INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State or Province of New Jersey and employed by Arkwright Mutual Insurance Company, Norwood, Massachusetts, have inspected the components described in this Owner's Data Report during the period 02-26-91 to 04-17-92 and state that to the best of my knowledge and belief the Owner has performed examinations and taken corrective measures described in this Owner's Data Report in accordance with the requirements of the ASME Code, Section XI.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the examinations and corrective measures described in this Owner's Data Report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date Nov. 6, 19 92

Factory Mutual Systems

James E. Lark
Inspector's Signature

Commissions

NJ 375 'I'
National Board, State,
Province and No.

II. DISCUSSION OF EXAMINATION ACTIVITIES

A. Pre-Outage Planning

In order to assure efficient performance of the ISI, SwRI devoted a concentrated effort toward planning and preparation for the examination activities. This included review of the Long-Term Plan, scrutinizing calibration block configuration to ensure that they met the requirements for the 83/S83 Code, and developing nuclear projects operating procedures (NPOPs) to supplement the PSE&G preapproved listing of NPOPs. These activities were accomplished by SwRI's NDE Engineering Section with assistance provided by the NDE Applications Section culminating in the development of an Examination Plan.

The Examination Plan was developed to provide the basis for the ISI in one comprehensive document. It provided a listing of the examination areas, copies of all procedures, calibration block drawings, weld identification figures, and other information pertinent to the performance of the ISI. The Examination Plan was reviewed and approved by PSE&G prior to SwRI's arrival on site.

B. Examination Procedures and Personnel

The NDE activities were performed using visual (VT), liquid penetrant (PT), magnetic particle (MT), and manual ultrasonic (UT) techniques. These examinations were performed by SwRI and Sonic Systems International, Inc. (SSI), personnel in accordance with SwRI NPOPs which had been preapproved by PSE&G personnel.

SwRI NPOPs were utilized to provide guidelines and controls for performance of onsite examination activities. This included procedures for weld joint identification marking, indication recording, records control, and resolution of indications.

NPOPs were written to conform to the requirements of the applicable sections of the ASME B&PV Code and SwRI's NQAPM. Some procedures were amended for specific examination purposes with Interim Change Notices (ICNs). These ICNs were necessary to incorporate examination requirements unique to particular examinations at Salem 1. The ICNs used are noted on the individual examination data sheets, NDE Summary Tables, and copies of the ICNs and their respective NPOPs are located in Appendix C.

SwRI examination personnel were certified in accordance with SwRI NPOP 2.0-NDES-101, "Nondestructive Examination Personnel Qualification and Certification," which incorporates the guidelines of SNT-TC-1A of the American Society for Nondestructive Testing and ASME Section XI B&PV Code. The SSI examiner was certified in accordance with SSI's training, qualification, and certification procedure number SSI-A-005, which complies with SNT-TC-1A. A copy of each examiner's certifications is included in Appendix E.

C. Examination Activities

This subsection provides a discussion of the various NDE activities performed during the ISI. SwRI examination data records are included in Volumes III through V. A discussion of equipment and materials used for these examinations appears in Subsections E and F of this section of the report.

1. VT Examinations

VT examinations were performed on the reactor pressure vessel (RPV) closure head washers.

2 PT Examinations

Solvent removable visible dye penetrant examinations were performed by SwRI personnel on selected Class 1 and Class 2 piping and support welds.

3 MT Examinations

Examination personnel used an AC yoke and dry particle MT to examine supports on the boron injection tank, and piping welds within the Feedwater and Main Steam Systems. Fluorescent MT was performed on the RPV studs and nuts.

4 UT Examinations

UT examinations were performed on Class 1 and Class 2 components, including vessels, studs, austenitic piping, ferritic piping, and dissimilar metal welds. Various techniques were used to perform the UT examinations, depending on material type and weld thickness. For typical vessel and piping examinations, the following techniques were used:

- a. A 0-degree lamination scan (UT0L) was used for detection of laminar reflectors which might affect interpretation of angle-beam results.
- b. A 0-degree scan (UT0W) was used for detection of reflectors in the weld for vessel examinations.
- c. For austenitic and ferritic welds, angle-beam scans of 45 degrees (UT45) were used for detection of reflectors oriented parallel to the weld.
- d. Angle-beam scans, using 45-degree search units directed parallel to the weld (UT45T) were used on piping for detection of reflectors oriented transverse to the weld. For vessel examinations, 45- and 60-degree scans parallel to the weld were performed for detection of transverse reflectors.
- e. For selected dissimilar metal welds, augmented 45-degree refracted longitudinal examinations were also used for detection of reflectors oriented parallel and transverse to the weld.

5 TDAS Examinations

Thickness measurements were recorded using SwRI's Thickness Data Acquisition System (TDAS) at 58 pipe locations to identify possible corrosion/erosion. Results of the TDAS examinations in the form of thickness plots for each fitting were reported to PSE&G personnel during the examinations, and a separate summary report was submitted.

6 Special Examinations

At the request of PSE&G, two special series of examinations were performed:

- a. In response to Nuclear Regulatory Commission (NRC) Bulletin 76-06, "Stress Corrosion Cracks in Stagnant, Low Pressure Piping Containing Boric Acid Solution at PWR's," one 14-inch Residual Heat Removal, two 10-inch Safety Injection, one 8-inch Safety Injection, seven 6-inch Safety Injection, and three 4-inch Safety Injection welds were examined.

- b. In response to NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," seventeen 1.5-inch Safety Injection and two 2-inch and two 3-inch Chemical and Volume Control welds were examined with UT for the third consecutive outage.

D. Examination Results

PT examinations revealed the following indications:

<u>Area</u>	<u>Description</u>	<u>Resolution</u>	<u>CNF No.</u>
2-SJ-1137-12 Pipe to Valve	7/32-inch linear indication on base material of pipe	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-1
2-SJ-2160-10 Elbow to Pipe	Two linear indications, one 3/8-inch, one 9/32-inch on base material of elbow	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-2
2-SJ-2160-9 Pipe to Elbow	1/8-inch linear indication on base material of elbow	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-3
2-CV-2147-14 Pipe to Elbow	3/8-inch linear indication on base material of elbow	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-4
4-SJ-2150-14 Pipe to Valve	3/16-inch round indication 1/2 inch from fusion line on pipe side of weld	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-5
6-PR-1103-12 Elbow to Flange	Four linear indications, one 3/4 inch, one 1/8 inch and two 3/16 inch on Flange side 1/2 inch from weld centerline.	Reexamination after mechanical removal revealed no recordable indications.	SAM 1-6, 1-6A, 1-6B, 1-6C
6-PR-1105-1 Nozzle to Safe End	3/16-inch linear indication on nozzle side, 1/4 inch from weld centerline	Indication is allowable by Code. Accepted "as is" by PSE&G personnel.	SAM 1-7
1-1/2-SJ-1132-40 Valve to Pipe	1/4-inch linear indication on base material of valve	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-8
6-SJ-1141-1 Reducer to Elbow	9-inch linear indication on weld	Reexamination after cosmetic buffing revealed no recordable indications.	SAM 1-9

MT examinations revealed the following indications:

<u>Area</u>	<u>Description</u>	<u>Resolution</u>	<u>CNF No.</u>
34-MS-2131-1PL-19 Pipe Lug	1/4-inch linear indication on lug side	Indication is allowable by Code. Accepted "as is" by PSE&G personnel.	SAM 1-10
34-MS-2121-2 Pipe to Valve	1-inch linear indication at fusion line on valve	Reexamination after mechanical removal revealed no recordable indications.	SAM 1-11, 1-11A, 1-11B
34-MS-2141-2 Pipe to Valve	13/16-inch linear indication outside Code-required examination area on valve body	PSE&G personnel forwarded to their Engineering for disposition. Seat 92-322.	SAM 1-12

VT and UT examinations revealed no reportable indications.

At the request of PSE&G personnel, each examination area was checked to ensure that the appropriate identification number and L_0 was etched or stamped on it. PSE&G-developed forms were used to report variances from the requirement.

Copies of the CNFs are included in Appendix G.

E. Equipment

Various NDE equipment was used during the ISI to perform the examinations of the selected components. The major equipment items are discussed below.

1. UT Instruments

Sonic 136 and FTS Mark I UT instruments were used for the NDE activities. To assure proper instrument linearity and operation, SwRI certified each instrument prior to use at Salem Unit 1, as specified in the NQAPM. Copies of certifications for those instruments utilized at Salem Unit 1 are contained in Appendix F. In addition, instrument linearity measurements were taken before and after completion of site activities as specified in SwRI NPOP SwRI-NDE5, "Ultrasonic Linearity Measurements."

2. Transducers

Various brands, sizes, and frequencies of UT transducers were used to perform the examinations. A transducer frequency of 1.5, 2.25, 5.0, or 10.0 MHz was used as specified in the applicable NPOP.

Prior to use at Salem 1, each transducer was tested and certified to be within SwRI's acceptance standards. This certification is performed within 12 months of use to verify that each transducer is performing within standards. A copy of each transducer's certification is contained in Appendix F.

Information on the actual transducer used for any specific examination may be determined by reviewing the data sheets and referenced calibration sheets in the field data volumes.

3. *Magnetic Particle Examination Equipment*

Hand-held alternating current magnetic particle yokes were used for MT examinations.

F. *Materials*

In addition to the equipment previously discussed, certain materials were required in the performance of the examinations. All materials contacting the examination surface (i.e., glycerine, pipe marking pencils, penetrants, etc.) were tested and certified to be within acceptable sulfur and halogen limits prior to use at Salem 1. Certifications for these materials are contained in Appendix F.

G. *UT Calibration Blocks*

Pipe and vessel calibration blocks were utilized to calibrate the UT instruments prior to examination of the selected welds. Drawings of the various calibration blocks used are contained in Appendix D.

III. SUMMARY OF EXAMINATIONS

This section provides a discussion of the field data records and a summary of the NDE activities performed at Salem 1.

A. Explanation of Field Data Records

The results of the examinations and calibrations performed by SwRI personnel were recorded on standard SwRI forms. Copies of these completed documents constitute Volumes III through V of this report. The original records will be retained in the SwRI Records Vault.

The field data records for each weld or area are assembled in a package preceded by a summary sheet. The examination areas and summary sheet numbers correspond to those listed in the Nondestructive Examination Summary Table provided in Subsection B below.

The instruments used in performing UT examinations were calibrated prior to use, then verified again at specified intervals during the examinations and upon completion of the examinations. The calibration parameters were recorded on the appropriate calibration record sheet as specified in the applicable NPOP. The documented calibration and calibration verification provide immediate assurance that the examinations were performed using properly calibrated instruments.

The results of the examinations were recorded on the applicable data record sheets as specified in the Examination Plan. The information documented on these forms describes the parameters associated with those indications which were greater than the recording levels specified in the applicable NPOP.

The size, location, and nature of reflectors were determined by analyzing the indication parameters recorded on the forms described above. The analysis is documented on SwRI Indication Resolution Record Sheets, which are included as part of each data package, when required.

B. Nondestructive Examination Summary Table

The following section is the Nondestructive Examination Summary Table (Summary Table). The Summary Table lists the areas that were examined by SwRI personnel during this ISI. See Figure 1 for an explanation of the Summary Table format. Weld identification drawings depicting the location of Class 1 and Class 2 components are contained in Appendices A and B, respectively.

DATE:
REVISION:

SALEM NUCLEAR POWER STATION UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, SECOND PERIOD, FIRST OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE:

REACTOR PRESSURE VESSEL CLOSURE HEAD

SUMMARY EXAMINATION AREA		ASME	EXAM	PROCEDURE	REMARKS	
NUMBER	IDENTIFICATION	SEC. XI CATGY ITEM NO	METHOD		E O E C N R	**CALIBRATION BLOCK**
<p>The remarks column is used to describe any pertinent or unique features of the examination such as limitations, results, CNFs, etc. Ultrasonic calibration blocks are also listed in this column.</p> <p>The results of the examination are indicated in these columns. The absence of recordable indications is shown by an "X" in the "NOEC" column. The presence of ultrasonic indications shown to be the result of a geometric feature of the examination area is indicated by an "X" in the "GEOM" column. The presence of nongeometric indications is indicated by an "X" in the "OTHER" column, and an explanation of the nature of each nongeometric indication is contained in the "REMARKS" column.</p> <p>This column lists the applicable SwRI nuclear projects operating procedure used for the examination.</p> <p>The NDE method used during the examination is listed in this column.</p> <p>The ASME SECTION XI ITEM NO. and CATEGORY of the examination area are listed in this column.</p> <p>Each examination area is listed in this column. Details of the weld identification system are contained in Appendices A and B.</p> <p>This column references the examination summary sheet which serves as a cover sheet for the data package and lists the data record numbers, the examiners, and any pertinent remarks.</p>						

Figure 1. Explanation of Summary Table Format

DATE: 08/27/92
REVISION: 0

SALEM NUCLEAR POWER STATION UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, SECOND PERIOD, FIRST OUTAGE (1992)
CLASS 1 COMPLETED COMPONENTS

PAGE: 16

PRESSURE RELIEF SYSTEM

SUMMARY EXAMINATION AREA		ASME			N	O	REMARKS
		SEC. XI			O	G	
		CATGY	EXAM	PROCEDURE	R	E	
NUMBER	IDENTIFICATION	ITEM NO	METHOD		E	O	
					C	M	**CALIBRATION BLOCK**

<u>6-PR-1105 (FIG NO A-13)</u>							
022200	6-PR-1105-1	B-F	PT	PSE-PT1/0	-	-	PT INDICATION IS CODE ALLOWABLE. SEE
	NOZZLE TO SAFE-END	85.130	UT45	PSE-UT31/1	X	-	CNF SAM1-7. ACCEPTED "AS IS" BY PSE&G
			UT45T		X	-	PERSONNEL.
			UT45RL	PSE-UT31/1	X	-	
			UT45RLT		X	-	**6-SS-160-.764-25-SAM**
022300	6-PR-1105-2	B-J	PT	PSE-PT1/0	X	-	UT45 INDICATION IS A GEOMETRIC REFLECTOR
	SAFE-END TO ELBOW	89.11	UT45	PSE-UT31/1	-	X	FROM THE WELD ROOT. NO UT FROM THE
			UT45T		X	-	UPSTREAM SIDE DUE TO SAFE-END
			UT45TAM		X	-	CONFIGURATION. OBTAINED 100% OF CODE
							REQUIRED COVERAGE.
							6-SS-160-.764-25-SAM
02200	6-PR-1105-11	B-J	PT	PSE-PT1/0	X	-	PT PERFORMED AT REQUEST OF PSE&G AS PART
	ELBOW TO FLANGE	89.11					OF AN EXPANDED SCOPE DUE TO INDICATIONS
							ON 6-PR-1103-12.
							6-SS-160-.764-25-SAM
<u>6-PR-1104 (FIG NO A-14)</u>							
024400	6-PR-1104-11	B-J	PT	PSE-PT1/0	X	-	PT PERFORMED AT REQUEST OF PSE&G AS PART
	ELBOW TO FLANGE	89.11					OF AN <u>EXPANDED SCOPE</u> DUE TO INDICATIONS
							ON 6-PR-1103-12.
							6-SS-160-.764-25-SAM
<u>6-PR-1103 (FIG NO A-15)</u>							
025700	6-PR-1103-12	B-J	PT	PSE-PT1/0	-	-	PT EXAM REVEALED ONE INDICATION WHICH
	ELBOW TO FLANGE	89.11	PT		X	-	EXCEEDED CODE ALLOWABLE DIMENSION.
			UT45	PSE-UT31/2	X	-	REEXAMINATION AFTER SURFACE PREPARATION
			UT45T		X	-	REVEALED NO RECORDABLE INDICATIONS. SEE
			UT45TAM		X	-	CNF SAM1-6, 6A, 6B AND 6C.
							6-SS-160-.764-25-SAM

Note:

DATE: 08/27/92
REVISION: 0

SALEM NUCLEAR POWER STATION UNIT 1
SUMMARY OF NONDESTRUCTIVE EXAMINATIONS
SECOND INTERVAL, SECOND PERIOD, FIRST OUTAGE (1992)
CLASS 2 COMPLETED COMPONENTS

PAGE: 4

MAIN STEAM SYSTEM

SUMMARY EXAMINATION AREA		ASME			N	O	
NUMBER IDENTIFICATION		SEC. XI	CATGY	EXAM	O	G	T
		ITEM NO	METHOD	PROCEDURE	R	E	H
					E	O	E
					C	H	R
							REMARKS
							CALIBRATION BLOCK
.....							
<u>30-MS-2131 (FIG NO B-26)</u>							
236440	30-MS-2131-8LD-O	C-F-2	MT	PSE-MT1/0	X	-	EXAMINED 12" OF WELD LENGTH.
	LONGITUDINAL	CS.52	UT45	PSE-UT41/1	X	-	
			UT45T		X	-	
							PL-1.5-CS-65-SAM
<u>8-MS-2141 (FIG NO B-29)</u>							
242000	8-MS-2141-1	C-F-2	MT	PSE-MT1/0	X	-	LIMITED MT EXAM DUE TO SURROUNDING
	WELDOLET TO ELBOW	CS.51	UT45	PSE-UT41/1	X	-	PIPING AND SUPPORT STRUCTURES. EXAMINE
			UT45T		X	-	65% OF THE REQUIRED AREA. NO UT
							UPSTREAM.
							B-CS-160-.906-46-SAM
<u>8-MS-2124 (FIG NO B-31)</u>							
243500	8-MS-2124-2	C-F-2	MT	PSE-MT1/0	X	-	LIMITED MT EXAM DUE TO SURROUNDING
	ELBOW TO VALVE	CS.51	UTOL	PSE-UT49/1	X	-	PIPING AND SUPPORT STRUCTURES. EXAMINE
			UT45	PSE-UT41/1	X	-	56% OF THE REQUIRED AREA.
			UT45T		X	-	
							B-CS-160-.906-46-SAM

*no relief
retained*

APPENDIX G
CUSTOMER NOTIFICATION FORMS

Table of Contents (Cont'd)

<u>CNF No.</u>	<u>Area</u>	<u>Indication(s)</u>	<u>Opened</u>	<u>Closed</u>
SAM 1-11, 11A, and 11B	34-MS-2121-2 Pipe to Valve	1-inch linear indication at fusion line on valve	5/13/92	5/19/92
SAM 1-12	34-MS-2141-2 Pipe to Valve	13/16-inch linear indication outside Code-required examination area on valve body	5/16/92	5/19/92

SOUTHWEST RESEARCH INSTITUTE
CUSTOMER NOTIFICATION FORM

Project No. 17-4823 Site: SALEM 1 CNF Serial No. SAM 1-12

Examination Area: 34-MS-2141-2 Procedure No. PSE-MT 1

PART I - SwRI Finding

Comments: MT EXAMINATION RECORDED ON DATA RECORD 120013
REVEALED AN INDICATION THAT IS OUTSIDE THE CODE
REQUIRED EXAMINATION AREA. THE INDICATION IS LOCATED
ON THE ADJACENT VALVE BODY (14-MS-167). INDICATION
IS 1 3/16" IN LENGTH

Attachments: DATA RECORDS 120013 AND 120074

SwRI Signature: Mickey T. Tene Date: 5/16/92

PART II - CNF Received By

Customer Signature: Wayne Dealing Date: 5/16/92

PART III - Disposition by Customer

Comments: THE INDICATION IS LOCATED OUTSIDE THE ASME XI CODE
REQUIRED EXAMINATION AREA ON A NUCLEAR CLASS II VALVE CASTING.
ASME SECTION III NC (CAST PRODUCTS) NC-2570 HAS NO REQUIREMENTS
OR ACCEPTANCE STANDARDS FOR MT EXAMINATION OF THE CASTING
SURFACES. A SYSTEM ENGINEERING ASSISTANCE TRANSMITTAL HAS BEEN WRITTEN
FOR ENGINEERING EVALUATION OF DISPOSITION OF INDICATION

Customer Signature: Wayne Dealing Date: 5/17/92

PART IV - Reexamination

Comments: REEXAMINATION AFTER MECHANICAL REMOVAL OF .205
REVEALED THAT THE INDICATION STILL EXISTS.
T BEFORE = 2.552, T AFTER = 2.347
EXAM PERFORMED AT SITE REQUEST PER DISPOSITION TO 921009017
SEAT 92-322

Attachments: DATA RECORD 120015

SwRI Signature: Mickey T. Tene Date: 5/19/92

PART V - CNF Closed By

Customer Signature: Michael D. [Signature] Date: 5/19/92

DISPOSITION TO 921009017
SEAT 92-322

14MS167 LINEAR INDICATION IN AREA OF PIPE TO VALVE WELD

THIS DISPOSITION WILL ADDRESS THE REPAIR OF 14MS167 VALVE. THE VALVES ARE PART OF THE PSEG PIPE SPEC. 1S12A, WHICH IS SAFETY RELATED, SEISMIC I NUCLEAR II AND QA REQUIRED. THE MAINT. REQUIRED TO REMOVE THIS INDICATION DOES NOT REQUIRE A CODE JOB PACKAGE.

1. LOCATE AND CLEARLY DEFINE THE AREA TO BE WORKED.
2. UT AREA FOR INITIAL WALL THICKNESS
3. USE MECHANICAL MEANS TO REMOVE INDICATION. DO NOT REMOVE MORE THAN 0.400 WITH OUT SYSTEM ENG APPROVAL.
4. NDE AS REQUIRED TO INSURE THAT THE INDICATION HAS BEEN REMOVED.
5. BY USING MECHANICAL MEANS CONTOUR THE AREA TO A 3/1 SLOPE.
6. FINAL NDE REQUIRED AFTER ALL MECHANICAL WORK HAS BEEN COMPLETED TO INSURE THAT THE INDICATION HAS BEEN REMOVED AND THE SURROUNDING AREA IS FREE OF DEFECTS.
7. FORWARD AS FOUND AND FINAL WALL THICKNESS TO SYSTEM ENG FOR DISPOSITION OF THE DR.

TESTING

1. PERFORM A MT OR PT EXAMS AFTER THE REMOVAL OF INDICATION IN ACCORDANCE WITH ANSI B31.7.
2. PERFORM A UT FOR AS FOUND AND AS LEFT WALL THICKNESS IN THE AREA OF THE INDICATION. ACCORDANCE WITH ANSI B31.7.

BD/Ker

5/18/92

Index of NIS-2 and NR-1 Forms

I. NIS-2 Forms

<u>Component</u>	<u>Brief Description</u>
1AF128	Replaced spring washer, spring and nozzle
11MS15	Machined nozzle
11SJ39	Replaced disk insert and spindle assembly
1SJ67	Replaced nozzle, disk and bellows

II. NR-1 Forms

<u>Code Job #</u>	<u>Brief Description</u>
P90-034	Replace spools 1099XA and 100X-A to accomodate new transmitters
P90-036	Addition of vent valves to RHR system
P90-039	Add new pipe supports to RHR system
P90-212	Replaced studs on valve 1SJ5
P90-313	Replaced 11SJ48 valve and bolting in kind
P90-316	Replaced 12CS10 valve, Mark # FA-17 with Mark # FA-131
P90-338	Installed new bonnet, studs and nuts in valve 11CS36
P90-339	Installed new bonnet, studs and nuts in valve 12CS36
P91-067	Replaced 11 and 12 FCU penetration piping with piping of new design and material
P91-068	Replaced 11 and 12 FCU penetration piping hangers and supports to accommodate new design
P91-071	Replaced 13, 14 and 15 FCU penetration piping with piping of new design and material
P91-072	Replaced 13, 14 and 15 FCU penetration piping hangers and supports to accommodate new design
P92-010	Replaced valve 11GB3 with new design and modified associated supports
P92-011	Replaced valve 12GB3 with new design
P92-012	Replaced valve 13GB3 with new design and modified associated supports
P92-013	Replaced valve 14GB3 with new design
S91-074	Replaced snubber 1C-CVC-SN901, S/N 321 with S/N 26130
S91-075	Replaced snubber 1S-14FW-SN013B, S/N 476 with S/N 30359
S91-076	Replaced snubber 1S-13FW-SN015B, S/N 389 with S/N 26131
S91-084	Replaced spool piece elbow 1MS-4037B in kind
S91-111	Replaced bonnet studs and nuts on valve 1CV284
S91-122	Repaired pitting on # 11 Steam Generator secondary handhole by machining
S91-167	Performed weld repair to pipe spool RD-1-SW-259
S91-238	Replaced bolting in flow orifice 1S12A
S91-256	Replaced bolting on valve 1CV7
S91-265	Replaced # 11 Boric Acid Transfer Pump mechanical seal
S91-280	Replaced # 12 Boric Acid Transfer Pump mechanical seal

S91-287 Weld repaired 15SW58 valve body and replaced flange bolting
 S91-306 Replaced cage, plug and stem assembly on valve 1PS1
 S92-097 Performed weld repair on 15SW72 valve body
 S92-155 Replaced (8) bonnet studs and nuts and (16) flange nuts and (8) studs on valve 1DR29
 S92-168 Performed weld repair of airlock personnel hatch hinge
 S92-179 Replaced bolting on valve 1CV171
 S92-188 Replaced snubber # 1A-SI-SN100A, S/N 450 with S/N 25872
 S92-190 Replaced snubber # 1S14-FWSN-15B, S/N 488 with S/N 27911
 S92-191 Replaced snubber # 1S11-FWSN-12B, S/N 440 with S/N 25893
 S92-192 Replaced snubber # 1S-11FW-SN012A, S/N 389 with S/N 25856
 S92-193 Replaced snubber # 1C-PR-SN007, S/N 623 with S/N 17676
 S92-200 Replaced stem and plug assembly on valve 12MS171
 S92-202 Performed weld repair on 12SW72 valve body
 S92-203 Performed weld repair on 12SW58 valve body and replaced valve bolting
 S92-215 Replaced disk in valve 1CV52
 S92-216 Replaced plug and stem assembly in valve 13GB4
 S92-218 Replaced plug and seat in kind in valve 11GB4
 S92-226 Replaced flanges and bolting in orifice 13MS200
 S92-227 Replaced bolting, plug and stem assembly in valve 14MS169. Hanger removed and reinstalled to facilitate maintenance
 S92-228 Performed weld repair to valve 13SW58
 S92-229 Performed weld repair on valve 13SW72
 S92-232 Replaced plug and stem assembly on valve 1PS3
 S92-235 Rewelded valve canopy after maintenance on valve 11SJ144
 S92-236 Rewelded valve canopy after maintenance on valve 14SJ17
 S92-241 Replaced bolting on valve 1SJ3
 S92-246 Rewelded valve canopy after inspection on valve 14SJ139
 S92-247 Replaced valve canopy by welding on valve 12SJ144
 S92-254 Replaced bolting on valve 11SJ34
 S92-256 Replaced bolting on valve 1CV7
 S92-264 Replaced helicoil in bolthole # 1 of # 11 Steam Generator
 S92-269 Replaced bolting in valve 11SJ55
 S92-270 Replaced primary manway bolting on # 14 Steam Generator hot leg
 S92-272 Replaced bolting and seal plate on valve 13SJ56
 S92-280 Replaced valve 14SJ144 with valve S/N CD-569
 S92-285 Replaced pipe clamp on support # 1C-PRH-179 due to corrosion
 S92-291 Replaced stem assembly and bolting on valve 1PR6
 S92-296 Performed weld repair on hanger 1C-CVCS-587 which was removed to facilitate maintenance
 S92-297 Replaced plug and stem assembly on valve 1CV75
 S92-300 Replaced rear bracket by welding on snubber 1C-PRSN-007

S92-301	Performed weld repair on seat and body and replaced bolting on valve 11BF22
S92-302	Performed weld repair on seat and body and replaced bolting on valve 12BF22
S92-303	Performed weld repair to body and seat on valve 13BF22
S92-304	Performed weld repair on seat and body and replaced bolting on valve 14BF22
S92-309	Replaced plug and stem assembly on valve 1CV3
S92-312	Replaced mechanical seal on 1SJE3 Safety Injection Pump
S92-313	Welded in new seat, replaced disk, valve bolting and flange bolting on valve 1FP148
S92-316	Performed weld repair to pipe support and replaced pipe guide on support 1C-CVCG-232
S92-317	Performed weld repair on flange face of 1SJE3 - # 12 Safety Injection Pump
S92-318	Replaced heat exchanger bolting with upgraded material on 1RHE3 heat exchanger
S92-325	Replaced bonnet and bolting on valve 1PS28
S92-329	Replaced cage assembly and performed weld repair on plug of valve 1PR2
S92-333	Performed weld repair to pipe spool between 1CS6 and 11CS44 valves
S92-335	Repaired welds on elbow spool 1CVC-15 that was removed to facilitate maintenance
S92-339	Weld 7/8" fillet weld in lieu of existing threads on yoke-bonnet for valve 14SJ54
S92-340	Replaced stem and wedge and enlarge wedge guides by machining on valve 1PR6
S92-344	Machined rupture disk seating surface on 1RCE11 - Pressurizer Relief Tank
S92-350	Replaced bolting on valve 12SW58
S92-352	Performed weld repair to reactor instrument tubing - 6522XD
S92-360	Replaced plug, stem assembly and bonnet on valve 22GB4
S92-362	Replaced plug and stem assembly and rewelded leak-off bore which was cut out to facilitate maintenance on valve 1CV55

III. Vendor NR-1 Forms

Work Order Brief Description

920627062	Repair # 11 Steam Generator nozzle and replace piping
920627064	Repair # 12 Steam Generator nozzle and replace piping
920627066	Repair # 13 Steam Generator nozzle and replace piping
920627067	Repair # 14 Steam Generator nozzle and replace piping

C-222

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MS-2209

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INTRODUCTION

This report presents the results of the Inservice Inspection (ISI) of the first outage, first period, third interval at the H. B. Robinson Steam Electric Plant, Unit 2. These examinations were conducted from March through June 1992. Nuclear Energy Services conducted these examinations in accordance with the program plan located under Tab C of this report. The program plan was derived from the Inservice Ten Year Inspection Plan. This plan is based on the requirements of the ASME Code, Section XI 1986 Edition and the Carolina Power & Light Specification No. HBR-ISI-1.

RESULTS

All scheduled nondestructive examinations performed during this ISI are documented on examination data sheets. Included on these examination data sheets are the component or system identification, examination method and procedure employed, unique data sheet number, and all applicable equipment settings and examination conditions. The data sheets, together with applicable isometric sketches, are presented in numerical order under Tab D of this report. In the event that a recordable indication was detected, all pertinent information was documented on the appropriate data sheet for evaluation. The evaluations of any recordable indications are located under Tab F of this report. A summary of indications is as follows:

CLASS I

Visual examinations were scheduled for the Reactor Coolant Pump Flywheels A, B, C, and 1 Spare. Flywheel "A" was deferred. The Spare Flywheel had a recordable indication that was dispositioned by Westinghouse. This indication is summarized under Tab F of this report.

CLASS II

Volumetric examinations revealed no flaw indications that exceeded the standards of ASME, Section XI IWA-3000. Included under Tab F are the final results of the Nuclear Regulatory Commission's concerns with Steam Generator "C" Nozzle Weld 1 (CP&L-217-1).

Surface examinations revealed several components with recordable indications. After engineering evaluation all but one was found to be acceptable. The one component support was unacceptable and declared inoperable. As a result, an additional number of examinations were performed in accordance with IWF-2430. The data for the additional workscope can be found in the Expanded Scope section under Tab D of this report. All indications are summarized under Tab F of this report.

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CLASS III

Visual examination of Class III component supports revealed several items with recordable indications. After engineering evaluation all were determined to be acceptable. All indications are summarized under Tab F of this report.

Sheet 1 of 119

FORM NIS-1 OWNER'S REPORT FOR INSERVICE INSPECTIONS

As required by the Provisions of the ASME Code Rules

1. Owner CAROLINA POWER & LIGHT COMPANY, 411 FAYETTEVILLE ST., RALEIGH, NC 27602
(Name and Address of Owner)
2. Plant H.B. ROBINSON, HWY. 151 & SC 23 HARTSVILLE, SC 29550
(Name and Address of Plant)
3. Plant Unit HRR2 4. Owner Certificate of Authorization (if required) N/A
5. Commercial Service Date 3/7/71 6. National Board Number for Unit N/A
7. Components Inspected

Component or Appurtenance	Manufacturer or Installer	Manufacturer or Installer Serial No.	State or Province No.	National Board No.
R.C.PUMP A	WESTINGHOUSE	CPLRCPCPC1	N/A	N/A
R.C.PUMP B	WESTINGHOUSE	CPLRCPCPC2	N/A	N/A
R.C.PUMP C	WESTINGHOUSE	CPLRCPCPC3	N/A	N/A
BORON INJ. TANK	SOUTH. FABRICATORS	0049	N/A	368
RHR H.X. A	WESTINGHOUSE	CPLACAHRS1	N/A	731
S.W. INJ. FILTER A	WESTINGHOUSE	CPLCHFLS11	N/A	929
LETDOWN H.X.	WESTINGHOUSE	CPLCHAMNR	N/A	685
PIPING	EBASCO SVCS. INC.	N/A	N/A	N/A
SUPPORTS	EBASCO SVCS. INC.	N/A	N/A	N/A
CHARGING PUMP 1A	UNION PUMP CO.	274158	N/A	N/A
PULS. DAMPENER A	JOHNSON CONTROLS	M-0391	N/A	N/A
STEAM GENERATOR A	WESTINGHOUSE	CPLRCPCSG1	N/A	N/A
STEAM GENERATOR C	WESTINGHOUSE	CPLRCPCSG3	N/A	N/A

Note: Supplemental sheets in form of lists, sketches, or drawings may be used, provided (1) size is 8 1/2 in x 11 in., (2) information in items 1 through 6 on this report is included on each sheet, and (3) each sheet is numbered and the number of sheets is recorded at the top of this form.

This form (E00029) may be obtained from the ASME Order Dept., Law Drive, Box 2300, Fairfield, NJ 07007-2300

FORM NIS-1 (Back)

8. Examination Dates 3/1/92 to 6/30/92 9. Inspection Interval from 2/19/92 to 2/19/0210. Applicable Editions of Section XI 1986 Addenda N/A

11. Abstract of Examinations. Include a list of examinations and a statement concerning status of work required for current interval. SEE TABS B & C VOLUME I

12. Abstract of Conditions Noted. SEE TAB B VOLUME I

13. Abstract of Corrective Measures Recommended and Taken. SEE TABS D & F VOLUME I & II

We certify that the statements made in this report are correct and the examinations and corrective measures taken conform to the rules of the ASME Code, Section XI.

Certificate of Authorization No. (if applicable) N/A Expiration Date N/ADate 9/18/92 19 92 Signed CPL By Ray H. Chalk
Owner

CERTIFICATE OF INSERVICE INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and the State or Province of TENNESSEE and employed by The Hartford Steam Boiler Insp. and Ins. Co. of Hartford, Connecticut have inspected the components described in this Owner's Report during the period 5-23-91 to 9-18-92, and state to the best of my knowledge and belief, the Owner has performed examinations and taken corrective measures described in this Owner's Report in accordance with the requirements of the ASME Code, Section XI.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the examinations and corrective measures described in the Owner's Report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Ramon C. Vallabene Commissions TN 971
Inspector's Signature National Board, State, Province, and Endorsement

Date 9-18 19 92

**H. B. ROBINSON UNIT 2
STEAM GENERATOR "C" FEEDWATER NOZZLE WELD 1
MAY 16, 1992**

To investigate low amplitude reflectors that had been recorded during previous examinations, an ultrasonic examination was conducted on May 13, 1992 by Mr. J. Coley, NRC Region 2, and Mr. A. Pennanen, Nuclear Energy Services.

This examination was conducted with a 45 degree shear wave (detection technique), RTD 70 degree refracted longitudinal wave (flaw tip detection technique) and a 30-70-70 mode converted longitudinal wave (ID flaw detection technique).

Calibration for the 45 degree and the 30-70-70 degree technique was performed on an ASME 18" OD calibration block. The 70 degree tip detection technique calibration was performed on a carbon steel 1" thick step block.

The 45 degree examination produced a reflector with a through wall dimension of .150" at 16" clockwise from top dead center of the nozzle facing the generator. A tip, .450" from the OD surface was also seen with the RTD 70 degree transducer (See plot on page 4).

To confirm the presence of this reported flaw, CP&L performed several additional examinations which are discussed as follows.

CP&L CONFIRMATORY EXAMINATIONS

On May 14, 1992, a radiographic examination of the subject area was performed with Iridium 192 using Kodak Type M film. 100 percent on the subject weld was radiographed.

A second series of ultrasonic examinations were conducted on the subject area by Mr. E. Black and Mr. A. Pennanen on May 14, 1992 using the CP&L NDE procedure for sizing planar flaws (NDEP-426). A 1.5" notched block was used for calibration. (See page 5 for sketch of block.)

Angles/Wave Modes/Techniques used were:

1. 45 degree shear wave (detection)
2. 45 degree shear wave (tip diffraction)
3. 45 degree refracted longitudinal wave (tip detection)
4. 60 degree refracted longitudinal wave (tip detection)
5. 70 degree refracted longitudinal wave RTD (tip detection)
6. 0 degree longitudinal (thickness)
7. 30-70-70 mode converted longitudinal wave technique (ID connected flaw)

The following is a discussion of the examinations conducted and the results obtained.

CP&L EXAMINATION RESULTS

1. **Radiography** - This examination revealed a counterbore at the same location as the UT reflector in question. Comparison with fabrication radiographs revealed no significant differences with respect to the counterbore indication.
1. **45 degree shear wave (Detection Technique)** - The reflector could be seen from both sides of the weld. From the nozzle side, an additional 20 dB was needed to see the reflector. The signals produced by this scan consisted of a corner reflector and two associated multiples (typical of a geometric type reflector).
2. **45 degree shear wave (Tip diffracted Technique)** - Scans from both sides of the reflector produced no tip signal.
3. **45 degree RL (Tip Detection Technique)** - Scans from both sides of the weld produced a reflector tip at .600" from the OD surface.
4. **60 degree RL (Tip Detection Technique)** - Scans from both sides of the weld produced no reflectors.
5. **70 degree RL (Tip Detection Technique)** - From the reducer side, this scan produced a reflector .450" from the OD surface. From the nozzle side, this scan produced a reflector .500" from the OD surface.
6. **0 degree longitudinal (Thickness Technique)** - A detailed profile of this weld at location L-18" is attached. A thickness of .645" exists at the location of the reflector.
7. **30-70-70** - This examination produced signals that could not be associated with wave modes normally produced by the WSY 70.

CONCLUSION

Based on the results obtained and recorded using the above examination techniques, this reflector does not appear to be associated with a crack. The characteristics exhibited are indicative of reflectors associated with geometric reflectors.

This conclusion is based on:

- Results obtained from the 45 degree shear wave, 45 degree RL, and 60 degree RL tests. See attached profile for full scale cross section of this indication (Attached Page 6).
- Radiography revealed a geometric condition (counterbore) in the same location as the reported ultrasonic reflector. Also, comparison with fabrication radiographs revealed no significant changes.

- The 70 degree RL results did not correspond with the results of the other tests. Using the 70 degree RL transducer, accurate depth measurements could not be obtained from the .600" deep remaining ligament and/or deeper notches on the sizing block used for calibration. Only the .200" and .400" notches were used. The .450" and .500" depths were outside the calibration range. (See attached sketch of the sizing block.)

As recommended by Mr. J. Coley on May 15, 1992, Mr. Stan Walker, EPRI NDE Center, performed a technical review of the ultrasonic methodologies used to confirm/deny the presence of a flaw. Mr. Walker also examined the subject weld using 45 degree shear, 70 degree longitudinal and 60 degree longitudinal wave modes and found no evidence of flaws within the subject weld. Mr. Walker will provide a trip report to Carolina Power & Light.

LEVEL II

1125

4 of 10
PAGE ____ OF ____DATA SHEET NO. _____
EXAM ITEM CPL 217-1
ISO DWG. NO. CPL 217 R

SKETCH SHEET

FOR INFORMATION ONLY

EXAMINER	<u>Cat P</u>	LEVEL	<u>III</u>	DATE	<u>5-13-92</u>
EXAMINER	<u>N/A</u>	LEVEL	<u>N/A</u>	DATE	<u>N/A</u>
REVIEWER	_____	LEVEL	_____	DATE	_____
REVIEWER	_____	DATE	_____		
REVIEWER	_____	DATE	_____

Caroline Power & Light Company

NDE DRAWING ATTACHMENT

PAGE 6 OF 6

PROJECT

JOB NO.

UNIT 1 ☐ 2 ☒ 3 ☐ 4 ☐

DATE 5/13/90

DRAWING

SYSTEM

S/C 'C'

LINE

FEEDWATER NOZZLE

WELD/ITEM NUMBER

WELD 1

SHEAR

MODE CONVERTED (BUTTED ON SURF.)

SHEAR REDIRECTED (WELD DAMP ON SURF.)

SCALE: 1" = 1"

CAROLINA POWER & LIGHT COMPANY

FORM QA UT-16A

ULTRASONIC CALIBRATION SHEET FOR PLANAR FLAW SIZING

REVISION 0

Plant: H.B. ROBINSON

Unit: 2

Date: 5/14/92

Sheet number:

Procedure: 426

Rev: 0

Couplant: UTRACEL

Batch No: 092091

Examiner: E.M. BLACK

Level: II

Calibration Block ID: CAL-UT-004

Pyrometer S/N:

Examiner: N/A

Level: N/A

Calibration Block Temp: AMBIENT

N/A

REFERENCE BLOCK

INSTRUMENT

TRANSDUCER

SIMULATOR BLOCK

ID: ~~_____~~Type: ~~_____~~

Material: N/A

Measured
Beam Angle: ~~_____~~☒ Digital ☐ Analog

Model: USK-7D

Serial No: 32810-894

Manufacturer: KB

Single ☒ Dual ☐

Serial No: F26110 M14135

Frequency: 2.25 Mhz

Size: 1/2"

Manufacturer: KBA

ID: ~~_____~~Reflector Type: ~~_____~~CE-2: ~~_____~~ Div's N/APulse Separation: ~~_____~~ Depth: ~~_____~~

INSTRUMENT SETTINGS

CALIBRATION

METHOD

CABLES

Reject OFF

Damping LOW

AP 5-14-92 Delay 13.9 μ sVEL Mat + Gal 210.9 μ s/ms N/A

Frequency .5 - 7 M

Gain/Coarse 57 dB

RANGE Gain/Time 2.5"

AP 5-18-92 Filter FIXED

Pulse/Echo ☒Thru trans ☐Jack Used: T ☒ / R ☐

Rep Rate FIXED

remarks

S/C 'C' NOZZLE WELD #1

screen div's

☒ % thru wall☐ depth in inches

Wave Mode:

shear: ☒longitudinal: ☐bi-modal: ☐

Type Reflector

holes: ☐notches: ☒☐ PATT☐ HALT☐ MOST☒ SPOT☐ 30-70-70CE-2: ~~_____~~ Div'sMeasured
Beam Angle
45 degRG62 ☐RG58 ☐RG174 ☒

Length: 6'

Initial Calibration

Time: 1440

CAL CHECKS

Time

AP 5-18-92

1820

Initials

N/A

REVIEWED BY:

LEVEL:

DATE:

5-18

ANII REVIEW:

AP 5/20/92

M. Valladares

DATE:

6-8-92

10 3 14

CAROLINA POWER & LIGHT COMPANY				FORM QA UT-16A	
ULTRASONIC CALIBRATION SHEET FOR PLANAR FLAW SIZING				REVISION 0	
Plant: <u>H.B. ROBINSON</u>		Unit: <u>2</u>	Date: <u>5/14/92</u>	Sheet number:	
Procedure: <u>NDEP 426</u>		Rev: <u>0</u>	Couplant: <u>ULTRACEL</u>	Batch No: <u>092041</u>	
Examiner: <u>E.M. BLACK</u>		Level: <u>II</u>	Calibration Block ID: <u>CR-UT-009</u>	Pyrometer S/N.	
Examiner: <u>N/A</u>		Level: <u>N/A</u>	Calibration Block Temp: <u>AMB</u>	<u>N/A</u>	
REFERENCE BLOCK		INSTRUMENT		TRANSducer	
ID: <u>N/A</u>		☒ Digital ☐ Analog		Single ☐ Dual ☒	
Type: <u>N/A</u>		Model: <u>USK 7D</u>		Serial No: <u>45RL(NRC)</u>	
Material: <u>N/A</u>		Serial No: <u>32810-894</u>		Frequency: <u>1.5</u> Mhz	
Measured Beam Angle: <u>N/A</u>		Manufacturer: <u>KB</u>		Size: <u>.50 x .50</u>	
				Manufacturer: <u>SUSI</u>	
				Pulse Separation: <u>N/A</u> Depth: <u>N/A</u>	
INSTRUMENT SETTINGS		CALIBRATION		METHOD	
Reject <u>0</u>				Wave Mode:	
Damping <u>FIXED</u>				shear: ☐	
Delay <u>9.745</u>				longitudinal: ☒	
Mat'l Cal <u>7601/ms</u>				bi-modal: ☐	
Frequency <u>.5-7M</u>				Type Reflector	
Gain/Coarse <u>42.5dB</u>				holes: ☐	
Gain/Fine <u>2.5"</u>				notches: ☒	
Filter <u>FIXED</u>				Measured Beam Angle <u>45RL</u> deg	
Pulse/Echo ☐				☐ PATT	
Thru trans ☒				☒ HALT	
Jack Used: T <u>✓</u> / R <u>✓</u>		☐ MOST			
Rep Rate <u>FIXED</u>		☐ SPOT			
remarks <u>S/C 'C' NOZZLE WELD #1</u>		☐ 30-70-70			
REVIEWED BY: <u>AM</u>		DATE: <u>5-18</u>		CE-2: <u>N/A</u> Div's	
LEVEL: <u>III</u>		DATE: <u>5/20/92</u>		ANII REVIEW: <u>RP</u>	
				DATE: <u>6-8-92</u>	

C-235

5/14

CAROLINA POWER & LIGHT COMPANY				FORM QA UT-16A											
ULTRASONIC CALIBRATION SHEET FOR PLANAR FLAW SIZING				REVISION 0											
Plant: <u>H.B. ROBINSON</u>		Unit: <u>2</u>	Date: <u>5/14/92</u>		Sheet number:										
Procedure: <u>NDEP 426</u>		Rev: <u>0</u>	Couplant: <u>ULTRACEL</u>		Batch No: <u>092041</u>										
Examiner: <u>E.M. BUACK</u>		Level: <u>II</u>	Calibration Block ID: <u>CPL-UT-004</u>		Pyrometer S/N										
Examiner: <u>N/A</u>		Level: <u>N/A</u>	Calibration Block Temp: <u>AMB</u>		<u>N/A</u>										
REFERENCE BLOCK		INSTRUMENT		TRANSDUCER											
ID: _____		☐ digital ☒ analog		Single ☐ Dual ☒											
Type: _____		Model: <u>USK 7D</u>		Serial No: <u>1348(NRC)</u>											
Material: <u>N/A</u>		Serial No: <u>32810-894</u>		Frequency: <u>1.5</u> Mhz											
Measured Beam Angle: _____		Manufacturer: <u>K-B</u>		Size: <u>.37 x .37</u>											
				Manufacturer: <u>SUSI</u>											
SIMULATOR BLOCK															
ID: _____															
Reflector Type: _____															
CE-2: _____ Div's <u>N/A</u>															
Pulse Separation: _____ Depth _____															
INSTRUMENT SETTINGS		CALIBRATION		METHOD											
Reject <u>0</u>				Wave Mode:											
Damping <u>FIXED</u>				shear: ☐											
Delay <u>9.9 us</u>				longitudinal: ☒											
Vel Mat-L Cal <u>467.6 m/s</u>				bi-modal: ☐											
Frequency <u>.5-7 MHz</u>				Type Reflector											
Gain/Coarse <u>42.5 dB</u>				holes: ☐											
RANGE Gain/Fine <u>2.5"</u>				notches: ☒											
Filter <u>FIXED</u>				☐ PATT Measured Beam Angle <u>N/A</u> deg ☒ HALT <u>60° RL</u> ☐ MOST ☐ SPOT ☐ 30-70-70 CE-2: _____ Div's											
Pulse/Echo ☐				RG62 ☐ RG58 ☐ RG174 ☒ Length <u>6 Ft</u>											
Thru trans ☒				Initial Calibration Time: <u>1410</u>											
Jack Used: T ☒ / R ☒		CAL CHECKS													
Rep Rate <u>FIXED</u>		<table border="1" style="width:100%; border-collapse: collapse;"> <tr> <th>Time</th> <th>Initials</th> </tr> <tr> <td><u>1833</u></td> <td></td> </tr> <tr> <td></td> <td></td> </tr> <tr> <td></td> <td></td> </tr> <tr> <td></td> <td></td> </tr> <tr> <td></td> <td></td> </tr> </table>		Time	Initials	<u>1833</u>									
Time	Initials														
<u>1833</u>															
remarks <u>S/G 'C' NOZZLE WELD #1</u>															
REVIEWED BY: <u>[Signature]</u>		LEVEL: <u>III</u>													
DATE: <u>5-1-92</u>		ANI REVIEW: <u>[Signature]</u>													
DATE: <u>6-8-9</u>															

C-236

110 01 12

CAROLINA POWER & LIGHT COMPANY
ULTRASONIC CALIBRATION SHEET FOR PLANAR FLAW SIZING

FORM QA UT-16A
REVISION 0

Plant: H.B. ROBINSON Unit: 2 Date: 5/19/92 Sheet number: _____

Procedure: NDER 426 Rev: 0 Couplant: ULTRACEL Batch No: 092091

Examiner: E.M. BLAKE Level: II Calibration Block ID: CPL-UT-004 Pyrometer S/N: _____

Examiner: N/A Level: N/A Calibration Block Temp: AMBIENT N/A

REFERENCE BLOCK INSTRUMENT TRANSDUCER SIMULATOR BLOCK

ID: _____ Digital ☒ Analog ☐ Single ☒ Dual ☐ ID: H1-128097

Type: _____ Model: USK 7D Serial No: 50526-00022 Frequency: 2 MHz Reflector Type: 10% TD NOTCH

Material: N/A Serial No: 32810-894 Size: _____ Manufacturer: KBA WSY CE-2: 1.0 DIV'S

Measured Beam Angle: _____ Manufacturer: KB Pulse Separation: N/A Depth: N/A

INSTRUMENT SETTINGS CALIBRATION METHOD

Reject 0 Wave Mode: _____ shears: ☐ longitudinal: ☒ bl-model: ☐

Damping LOW Delay 8.5 us Type Reflector: ☐ holes: ☐ notches: ☒

Mat'l Cal 121.6 mm N/A Frequency 5-7 Initial Calibration Time: 1300

Gain/Coarse 51dB Filter N/A CAL CHECKS

Pulse/Echo ☒ 2.5 Thru trans ☐ Time 1840 Initials _____

Jack Used: T 1/R As thru wall W depth in inches CE-2 30-70-70 1840 N/A

Rep Rate FIXED As thru wall W depth in inches CE-2 30-70-70 1840 N/A

REMARKS S/C 'C' NOZZLE USED #1 As thru wall W depth in inches CE-2 30-70-70 1840 N/A

REVIEWED BY: CE-2 LEVEL: III DATE: 5-92 ANI REVIEW: 5/20/92 DATE: _____

_____ DATE: _____

_____ DATE: _____

CAROLINA POWER & LIGHT COMPANY						FORM QA UT-16A	
ULTRASONIC CALIBRATION SHEET FOR PLANAR FLAW SIZING						REVISION 0	
Plant: <u>H.B. ROBINSON</u>		Unit: <u>2</u>		Date: <u>5/14/92</u>		Sheet number:	
Procedure: <u>426</u>		Rev: <u>0</u>		Couplant: <u>ULTRAGEL</u>		Batch No: <u>092091</u>	
Examiner: <u>E.M. BLACK</u>		Level: <u>II</u>		Calibration Block ID: <u>CPL-UT-004</u>		Pyrometer S/N:	
Examiner: <u>N/A</u>		Level: <u>N/A</u>		Calibration Block Temp: <u>AMBIENT</u>		<u>N/A</u>	
REFERENCE BLOCK		INSTRUMENT		TRANSDUCER		SIMULATOR BLOCK	
ID: <u>CPL-004</u>		☒ Digital ☐ Analog		Single ☐ Dual ☒		ID: <u>CPL-004</u>	
Type: <u>NOTCHED</u>		Model: <u>USK-7D</u>		Serial No: <u>83-924</u>		Reflector Type: <u>NOTCH</u>	
Material: <u>C/S</u>		Serial No: <u>NDC-32810-84</u>		Frequency: <u>2MHz</u> Mhz		CE-2: <u>N/A</u> Div's	
Measured Beam Angle: <u>N/A</u>		Manufacturer: <u>K-B</u>		Size: <u>6 x 13</u>		Pulse Separation: <u>N/A</u> Depth <u>N/A</u>	
Manufacturer:		Manufacturer:		Manufacturer:		Pulse Separation:	
INSTRUMENT SETTINGS		CALIBRATION		METHOD		CABLES	
Reject	<u>OFF</u>			Wave Mode:		RG62 ☐	
Damping	<u>Fixed</u>			shear: ☐		RG58 ☐	
Delay	<u>9.7us</u>			longitudinal: ☒		RG174 ☒	
Mat'l Cal	<u>132.5%</u>			bi-modal: ☐		Length <u>6'</u>	
Frequency	<u>.5-7M</u>			Type Reflector		Initial Calibration	
Gain/Coarse	<u>60dB</u>			holes: ☐		Time: <u>1230</u>	
Gain/Fine	<u>N/A</u>	notches: ☒		CAL CHECKS			
Filter	<u>N/A</u>			Time		Initials	
Pulse/Echo	☒			☐ PATT Measured Beam Angle ☒ HALT <u>TO RL</u> deg ☐ MOST ☐ SPOT ☐ 30-70-70 CE-2: <u>N/A</u> Div's		1844	
Thru trans	☒						
Jack Used: T	☒ / R ☒						
Rep Rate	<u>FIXED</u>						
remarks	<u>S/G 'C' NOZZLE WELD # 1</u>						
REVIEWED BY:	LEVEL:		DATE:		ANII REVIEW:		DATE:
					<u>RBW</u> <u>5/20/92</u> <u>AM Belladonna</u>		<u>6-8-92</u>

C-238

THICKNESS CALIBRATION SHEET

PLANT/UNIT: H.B. ROBINSON

COMPONENT: 5/8" C' FW NOZZLE WELD
#1

PROCEDURE #: 426 REV. 0

CALIBRATION BLOCK #1 89-012 BLOCK THICKNESS .25 → 1.5"

2 _____ BLOCK THICKNESS _____

INSTRUMENT: US 14 7 D

CAL. DUE DATE: 12.5 92

SERIAL NUMBER: 32810-894

SWEEP: 230.7 "/ms

DELAY: 3.6 us

RANGE: 1.0"

TRANSDUCER FREQUENCY: 5.0

TRANSDUCER SERIAL NO.: H28181

TRANSDUCER BRAND: KBIA

TRANSDUCER SIZE: 1/4

TRANSDUCER ELEMENT: SINGLE OR DUAL

EACH MAJOR SCREEN DIVISION: .100"

[illegible]

COMMENTS: _____

GAIN = 53 dB

ZERO = 3.79 μ S (DIGITAL)

EXAMINERS I E.M. BLACK DATE 5/14/92 LEVEL II

2 N/A DATE N/A LEVEL N/A

REVIEWERS: Art Puma DATE 5-18-92

2 Richard B. Fisher DATE 5/20/92

3 Arbhadra DATE 6.8.92

ADDITIONAL SHEETS. YES NO

RADIOGRAPHIC NDE REPORT

PROJECT HBR JOB NO. N/A UNIT 1 ☐ 2 ☒ 3 ☐ 4 ☐

DRAWING N/A SYSTEM FW LINE _____ WELD/ITEM NUMBER N/A

DESIGN CLASS/CATEGORY II ACCEPTANCE STANDARD
☐ ASME III ☐ VIII ☐ B31.1 ☐ AWS D1.1 ☒ OTHER INFO.

MTL TYPE CLS MTL THICKNESS 1 1/2" O.D./LENGTH 18" SURFACE FINISH: ☐ AS-WELDED ☒ GROUND ☐ OTHER _____ NDE PROCEDURE NO. 101 REV 14

STAGE OF MFG. ☒ INFO ☐ REPAIR ☐ ROOT ☐ INTERMEDIATE ☐ FINAL JOINT DESIGN ☐ BRN ☐ BKS ☐ INS ☐ OPN. BT ☒ N/A ☐ SOCKET ☐ OTHER _____

MAKE <u>X-RAY</u> <u>N</u>		ISOTOPE ☒ IRIDIUM 192 ☐ COBALT 60		LEAD SCREENS	
KVP. <u>N/A</u> <u>A</u>	SIZE DIA. <u>.107"</u> LENGTH <u>.102"</u>	FRONT ☐ .005 ☒ .010		CENTER <u>N/A</u>	BACK ☐ .005 ☒ .010
FOCAL SPOT SIZE	CURIES <u>59</u>	RT STANDARD SET-UP (CHECK ONE)			
FILM MFG. <u>"M"</u> ASTM CLASS 1 ☒ 2 ☐	LOADED 1 ☐ 2 ☒ OTHER ☐
PPD. <u>18"</u>	EXPOSURE TIME MIN. <u>25</u> SEC.
PENETRATOR SIZE <u>12</u> MTL. <u>9/8</u> ASTM ☐ ASME ☒	SOURCE SIDE ☐ FILM SIDE ☒
SHIM MTL. <u>N/A</u> THICK	COLLIMATION YES ☒ NO ☐
RADIOGRAPHER <u>MVP / PETE TINGEN</u>	CERT. LEVEL <u>II</u>	RT COMPANY <u>CP&L</u>			
VIEWING SINGLE ☒ COMPOSITE ☐	FILM PROCESSING ☐ AUTOMATIC ☒ MANUAL	TEMPERATURE <u>68°F</u>			
OTHER (ATTACH SKETCH) ☐					

LOCATION MARKERS	DENSITY		DISCONTINUITIES																	REMARKS	
	SENSITIVITY	WELD	PENETRATOR	ACCEPT	REJECT	POROSITY	SLAG INCL.	CRACK	INCOMPLETE PENET.	INCOMPLETE FUSION	UNDERCUT	ROOT UNDERCUT	BURN-THRU	CRATER PIT/CRACK	ROOT CONVEXITY	ROOT CONCAVITY	TUNGSTEN	UNCOM. INSERT	SURFACE DEFECTS	ARTIFACTS	STATE "N.A.D." IF NO APPARENT DISCONTINUITIES
0-1	2T		✓																		<p>Station Markers Looking Towards STM. GEN.</p>
1-2			✓																		
2-3			✓																		
3-4			✓																		
4-0			✓																		
FOR INFO ONLY																					

INTERPRETER [Signature] CERTIFICATION LEVEL II FILM EVALUATION DATE 5/14/92

IF APPLICABLE CONTRACTOR REVIEWER _____ TITLE _____

FILM REVIEW DATE _____ COMMENTS _____

IF APPLICABLE CP&L QA REVIEW [Signature] DATE 5/14/92 IF APPLICABLE XRAY/CP&L DATE 6.8.92

**SUPPLEMENTAL REPORT FOR
H. B. ROBINSON UNIT 2
STEAM GENERATOR "C" FEEDWATER NOZZLE WELD 1
SUMMARY DATED MAY 16, 1992**

The purpose of this attachment is to evaluate the .450" from outside diameter (OD) signal produced on May 14, 1992 using RTD transducer, S/N 83-924 during inspection of FW nozzle weld 1.

Attached are full scale plots and profiles of transducer exit point relative to holes, notches and counterbore to illustrate amplitude and screen positions of the 70° refracted longitudinal (RL) and the 34° shear (S) beam angles from the dual element RTD transducer, S/N 83-924 (enclosed).

The exit point and beam angles were measured using a type II, carbon steel IIW block and verified on various side drilled holes as shown on pages 2 and 3. Reflected 70° RL signals can easily be seen from side drilled holes with depths of up to .600" from the OD surface with no other accompanying signals that could confuse the operator.

As shown on page 4, using a one inch thick notched block, each tip diffracted RL signal is accompanied by a trailing 34° corner reflected shear wave signal at screen position 10. With a thinner block or component, the shear signal would be closer in time to that of the tip diffracted signal which could confuse the operator. (Notice the amplitude difference between the longitudinal and shear signals.)

On page 5, using a carbon steel block close in thickness to that of the feedwater nozzle weld 1 at H. B. Robinson, a 70° RL and 34° S signal could be obtained from the inside diameter (ID) notch. Because the metal path of the corner reflected shear component is almost half that of the RL tip signal, the screen position of the 34° shear component appears almost in the same location as that of the longitudinal signal.

Page 6 shows the approximate RTD probe location during the May 14 examination relative to the counterbore. It is possible that the counterbore signal was the cause of the signal seen at .450" from the OD surface. This is based on the signal presentation from the .600" deep hole shown on page 2. The sweep positions of the shear and longitudinal components are too close to avoid possible operator confusion. Also, the short echo dynamic response noted from the .450" deep signal on May 14 was characteristic of a shear signal; longitudinal signals exhibit long echo dynamic responses.

According to the EPRI Text, UT Operator Training for Sizing IGSCC, "The primary intent when sizing with the high-angle, L-wave method, is to limit the penetration of the beam into the metal in order to avoid confusion from unidentified ultrasonic signals." (Module 3, page 11, 10/89 Edition.)

We conclude that for the feedwater nozzle in steam generator C, weld 1 at H. B. Robinson Unit 2, this transducer should be used for flaws that penetrate within .300" of the outside surface (3 screen divisions) to reduce the possibility of shear wave encroachment of the calibrated region. (5.2 screen divisions on page 2.) This will prevent possible misinterpretation of shear signals caused by internal weld geometry.

Signed: Carl R. Cannon Date: 3 June, 1992

SCREEN PRESENTATIONS RELATIVE TO TRANSDUCER PLACEMENT
FOR 70° REFRACTED LONGITUDINAL AND TRAILING 32° SHEAR COMPONENT

AT 52 dB

NOTE: THE SCREEN WAS CALIBRATED SO THAT EACH
MAJOR DIVISION = .100" DEPTH FROM D.D.
SURFACE - USING 70° RL.

SCREEN PRESENTATIONS RELATIVE TO TRANSDUCER PLACEMENT

At 52 dB

- ① 32° SHEAR COMPONENT PEAKS AT 1" IN FROM 70° L
- ② EACH SCREEN DIVISION = .100" FM OD. SURFACE.
- ③ REFLECTORS = $\frac{1}{16}$ " DIA. SIDE DRILLED HOLES

SCREEN PRESENTATIONS RELATIVE TO TRANSDUCER PLACEMENT

AT 52 dB

① REFLECTORS = NOTCHES

2. FOR L-WAVE: EACH SCREEN DIV = .100" FM O.D. SURFACE

SCREEN PRESENTATIONS RELATIVE TO TRANSDUCER PLACEMENT
FOR 70° REFRACTED LONGITUDINAL AND TRAILING 32° SHEAR
COMPONENT

At 58 dB

CP&L
Carolina Power & Light Company
NDE DRAWING ATTACHMENT

P6 of 6

PAGE OF

PROJECT H.B. ROBINSON JOB NO.

UNIT 1 ☐ 2 ☒ 3 ☐ 4 ☐

DATE 5/20/92

DRAWING <u>S/C 'C'</u>	SYSTEM <u>FEEDWATER NOZZLE</u>	LINE <u>TO EXPANDER</u>	WELD/ITEM NUMBER <u>WELD #1</u>
---------------------------	-----------------------------------	----------------------------	------------------------------------

← WELD TOE

②

①

①

②

TOOR

TOOR

NOZZLE

EXPANDER

34° SHEAR

① APPROXIMATE LOCATIONS OF PROBE DURING 5/14 EXAM.

② EXIT POINT LOCATION REQUIRED TO SEE FLAW TIP THAT WAS REPORTED 5/13.

H. B. ROBINSON UNIT 2
STEAM GENERATOR C FEED WATER
NOZZLE TO REDUCER WELD #1

PAGE 1 OF 3

INTRODUCTION:

PERFORMED BY EXAMINATION OF THIS
WELD AND ADJACENT BASE MATERIAL FROM
2 1/2" UPSTREAM OF WELD #2 TO 2 1/2"
DOWNSTREAM OF WELD #1 ON 5-13-92 WITH
JIM COLEY (NRC INSPECTOR). A 45° SHEAR
WAVE AXIAL EXAMINATION DETECTED AN
INDICATION ORIGINATING FROM THE UPSTREAM
COUNTER BORE AT 16° CLOCKWISE AND
TERMINATING AT 32 1/2° CLOCKWISE.

ON 5-14-92 THE INDICATION AREA WAS
REINVESTIGATED BY MYSELF AND E. M. BLACK
(CPL) TO VERIFY THE INDICATION AND ITS
SIZE.

EXAMINATION:

ON 5-13 FOLLOWING EXAMINATIONS WERE
PERFORMED.

- ① 45° SHEAR WAVE AXIAL (DETECTION TECHNIQUE)
EXAMINATION WAS PERFORMED AT 12 O'CLOCK
REFERENCE AND IN THE UPSTREAM AND DOWNSTREAM
DIRECTIONS. SINCE THE INDICATION DID NOT
EXCEED 50% DAC, SIZE THROUGH WALL WAS DETERM.
AT 50% OF MAX AMPLITUDE. LENGTH WAS
DETERMINED AT COMPLETE LOSS OF SIGNAL.

② AN ID CREEPING WAVE, USING A W54-K0-2, WAS UTILIZED TO VERIFY A FLAW BREAKING THE ID SURFACE. A SIGNAL WAS SEEN, HOWEVER, SINCE THE OA AND IA SURFACES WERE NOT PARALLEL THIS DETECTION/SIZING TECHNIQUE WOULD NOT PRODUCE VALID RESULTS.

③ AN RTD 70° RL (TIP DETECTION TECHNIQUE) WAS UTILIZED FROM THE REDUCED SIDE. A REFLECTION WAS RECEIVED AT 400" BELOW THE SURFACE.

④ A 0° LONGITUDINAL (THICKNESS TECHNIQUE) WAS UTILIZED TO DETERMINE A THICKNESS PROFILE FROM PLOTTING THE INDICATION.

ON 5-14 THE FOLLOWING EXAMINATIONS WERE PERFORMED

① 45 SHEAR WAVE COULD EASILY DETECT THE INDICATION FROM REDUCED. AN ADDITIONAL 20 DB WAS REQUIRED TO OBSERVE A 20% FSH REFLECT WITH THE SEARCH UNIT ON THE WELD CROWN.

② 45 SHEAR WAVE (TIP DIFFRACTION TECHNIQUE) PRODUCED A TIP SIGNAL FROM THE REDUCED SIDE OF .100" TO .150" THROUGH WALL FROM THE ID SURFACE.

- (3) 45 DEGREE RL (TIP DETECTION TECHNIQUE) PRODUCED A SIGNAL AT .600" FROM THE O.D. SURFACE.
- (4) 60 DEGREE RL (TIP DETECTION TECHNIQUE) PRODUCES NO MEANINGFUL RESULTS.
- (5) 70 DEGREE RL RTD (TIP DETECTION TECHNIQUE) PRODUCED A REFLECTOR AT .450" FROM THE OD SURFACE WHEN SCANNED FROM THE REDUCER SIDE. WHEN SCANNED FROM THE WELD CROWN THE REFLECTOR WAS DETECTED AT .500" FROM THE O.D. SURFACE.
- (6) 0 DEGREE LONGITUDINAL (THICKNESS TECHNIQUE) A THICKNESS OF .645" WAS DETECTED AT THE WELD EDGE IN THE AREA OF THE INDICATION. THICKNESS INCREASED TO OVER 1" ON THE REDUCER AND WERE VARIABLE ON THE WELD CROWN TO .800". A COUNTERBORE WAS DETECTED BELOW THE WELD EDGE.
- (7) A WSY 70-2 CREEPING WAVE (DETECTION/SIZING TECHNIQUE) WAS UTILIZED. REFLECTORS (MULTIPLE, INDICATED THE PRESENCE OF NON PARALLEL SURFACES ON THE REDUCER SIDE AND WOULD NOT GENERATE THE DESIRED CREEPING WAVE. A SCAN FROM THE WELD SIDE PRODUCED A FLAW BASE SIGNAL BUT COULD NOT DETECT REFLECTOR

FROM THE TOP / SIDE OF THE FLAW.

CONCLUSION :

THIS REFLECTOR APPEARS TO BE A SMALL CRACK ORIGINATING FROM THE JUNCTION OF COUNTERBORE AND TRANSITION SURFACES. THE DEPTH FROM THE ID SURFACE APPEARS TO BE $\approx .150"$. THIS CONCLUSION WAS BASED ON THE EXAMINATION RESULTS OF 5-13 AND 5-14. IN ADDITION THE CRACK IS CONSISTENT WITH THE LOCATION OF KNOWN CRACKS FOUND IN THIS AND OTHER PLANTS.

Celt Penner UT LEVEL
5-15-92

Nuclear

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(717) 948-8005

February 13, 1992
C311-92-2014

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

*Note:
see Appendix E.*

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Inservice Inspection (ISI) Data Reports

In accordance with 10CFR 50.55a(g) and TMI-1 Technical Specifications, enclosed are GPU Nuclear's Forms NIS-1, "Owner's Data Report for Inservice Inspection" and NIS-2 "Owner's Data Report for Repairs or Replacements."

Two NIS-1 Forms are enclosed. One NIS-1 Form covers ISI first interval examinations conducted between March 5, 1990 and November 15, 1991 which were conducted in accordance with the ASME Code, Section XI, 1974 Edition through Summer 1975 Addenda. The second NIS-1 Form covers ISI second interval examinations conducted between April 19, 1991 through November 15, 1991 in accordance with the ASME Code, Section XI, 1986 Edition with no Addenda. Table 1 of the NIS-1 provides an abstract of the examinations performed. Table 2 provides an abstract of the corrective measures recommended and taken. All conditions listed on Table 2, where Code acceptance criteria were exceeded, were determined to be operable upon inspection and evaluation. Repairs were performed as indicated in the table to meet Code requirements. These reports are being submitted in compliance with Subarticle IWA-6220 of the Code.

The enclosed NIS-2 reports cover those repairs or replacements where the documentation was completed between April 4, 1990 and January 10, 1992. In accordance with Subarticle IWA-6220, only the NIS-2 reports describing Class 1 or 2 component repairs or replacements need be submitted.

9202200044 920213
PDR ADOCK 05000289
Q PDR

GPU Nuclear Corporation is a subsidiary of General Public Utilities Corporation

AD47 11

NIS-1 OWNERS DATA REPORT FOR INSERVICE INSPECTION

DATE: 02/03/92

PAGE: 0001

1. OWNER: Metropolitan Edison Company, 2800 Pottsville Pike, Reading, Pennsylvania
2. PLANT: Three Mile Island Nuclear Generating Station
Route 441 South, Middletown, Pennsylvania
3. PLANT UNIT: 1
4. OWNER CERTIFICATION OF AUTHORIZATION (if required): None
5. COMMERCIAL SERVICE DATE: 09/02/74
6. NATIONAL BOARD NUMBER FOR UNIT: Reactor Vessel N-109

INTERVAL 1

OUTAGE 9R
TABLE 1

ASME CAT	ASME ITEM NO.	IDENTIFICATION TAG NUMBER SUFFIX	CODE	DESCRIPTION	SYST	NDE METH	EXAM DATE/ REASON	FROM DIAM./ TO DIAM.	REMARKS
B-A	B1.1	RC-T-0001 A1-205-2	WELD	SHELL COURSE REPAIR WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF REPAIR AREA.
B-A	B1.1	RC-T-0001 RV-0013	WELD	SHELL TO SHELL CIRC WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF CIRCUMFERENCE.
B-A	B1.1	RC-T-0001 RV-0014	WELD	SHELL TO SHELL CIRC WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF CIRCUMFERENCE.
B-A	B1.1	RC-T-0001 RV-0018L	WELD	SHELL TO SHELL LONG WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF LENGTH.
B-A	B1.1	RC-T-0001 RV-0019 L	WELD	SHELL TO SHELL LONG WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF LENGTH.
B-A	B1.1	RC-T-0001 RV-0020 L	WELD	SHELL TO SHELL LONG WELD	RC	UT	103091 ISI	189.000	LIMITED DUE TO CORF GUIDE LUGS.
B-A	B1.1	RC-T-0001 RV-0021 L	WELD	SHELL TO SHELL LONG WELD	RC	UT	103091 ISI	189.000	EXAMINED 100% OF ACCESSIBLE LENGTH.
B-B	B1.2	RC-T-0001 RV-0012	WELD	SHELL TO SHELL CIRC WELD	RC	UT	103091 ISI	192.000	LIMITED DUE TO CORE GUIDE LUGS AND FLOW VANES. EXAMINED 5% OF CIRC.
B-B	B1.2	RC-T-0001 RV-0015	WELD	TRANSITION RING TO HEAD WELD	RC	UT	103091 ISI	182.000	EXAMINED 5% OF CIRCUMFERENCE.
B-B	B1.2	RC-T-0001 RV-0016	WELD	TUBESHEET TO HEAD WELD	RC	UT	111291 ISI	137.000	LIMITED DUE TO OBSTRUCTION
B-B	B3.1	RC-H-0001B SG0012	WELD	FLANGE TO SHELL WELD	RC	UT	103091 ISI	137.000	EXAMINED 5% OF CIRC LENGTH
B-C	B1.3	RC-T-0001 RV-0011	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER AND ADJACENT NOZZLES. EXAMINED 100% OF CIRC.
B-D	B1.4	RC-T-0001 RV-0001N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0002N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0003N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0004N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0005N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0006N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER.
B-D	B1.4	RC-T-0001 RV-0007N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER AND VENTURI.
B-D	B1.4	RC-T-0001 RV-0008N	WELD	NOZZLE TO SHELL WELD AND INNER RADIUS	RC	UT	103091 ISI	192.000	LIMITED DUE TO TAPER AND VENTURI.
B-F	B1.6	RC-T-0001 RV-0009BM	WELD	SAFE END TO NOZZLE WELD	RC	UT	103091 ISI	15.620	LIMITED DUE TO TAPER.
B-F	B1.6	RC-T-0001 RV-0010BM	WELD	SAFE END TO NOZZLE WELD	RC	UT	103091 ISI	15.620	LIMITED DUE TO TAPER.

Not
quantified,
relict not
referenced.

C-252

1. OWNER: Metropolitan Edison Company, 2800 Pottsville Pike,
Reading, Pennsylvania
2. PLANT: Three Mile Island Nuclear Generating Station
Route 441 South, Middletown, Pennsylvania
3. PLANT UNIT: 1
4. OWNER CERTIFICATION OF AUTHORIZATION (if required): None
5. COMMERCIAL SERVICE DATE: 09/02/74
6. NATIONAL BOARD NUMBER FOR UNIT: Reactor Vessel N-109

OUTAGE 9R
TABLE 1

INTERVAL 1

ASME CAT	ASME ITEM NO.	IDENTIFICATION		DESCRIPTION	SYST	NDE METH	EXAM DATE/ REASON	FROM DIAM./ TO DIAM.	REMARKS
		TAG NUMBER SUFFIX	CODE						
B-H	B3.7	RC-H-0001B		HEAD TO TRANSITION RING W	RC	MT	111291	146.000	
		SC-0013	WELD	ELD			ISI	146.000	
B-I-1	B1.14	RC-T-0001		VESSEL CLAD PATCH	RC	VIS	103091		
		RV-0003CL	INTER				ISI		
B-I-1	B1.14	RC-T-0001		VESSEL CLAD PATCH	RC	VIS	103091		
		RV-0006CL	INTER				ISI		
B-J	B4.5	CF-0012		PIPE TO TEE WELD	CF	UT	103091	14.000	
			WELD				ISI	14.000	
B-J	B4.5	RC-0087		PIPE TO NOZZLE WELD	RC	UT	103091	28.000	
			WELD				ISI	28.000	
B-N-1	B1.15	RC-T-0001		VESSEL INTERIOR	RC	VIS	103091		
			INTER				ISI		
B-N-3	B1.17	RC-T-0001		VESSEL CORE SUPPORT ASSEM	RC	VIS	103091		
		CSA	INTER	BLY			ISI		
C-D	C2.4	MS-0092BT		FLANGE BOLTING MS-V20D	MS	VIS	110191	6.000	
			FLANGE				ISI		
C-E-1	C2.05	DH-0019WA		INTEGRAL ATTACHMENT WELD	DH	PT	100291	14.000	
			WELD				ISI		

C-253

NIS-1 OWNERS DATA REPORT FOR INSERVICE INSPECTION

DATE: 02/03/92

PAGE: 0001

1. OWNER: Metropolitan Edison Company, 2800 Pottsville Pike,
Reading, Pennsylvania
2. PLANT: Three Mile Island Nuclear Generating Station
Route 441 South, Middletown, Pennsylvania
3. PLANT UNIT: 1
4. OWNER CERTIFICATION OF AUTHORIZATION (if required): None
5. COMMERCIAL SERVICE DATE: 09/02/74
6. NATIONAL BOARD NUMBER FOR UNIT: Reactor Vessel N-109

OUTAGE 9R
TABLE 2

INTERVAL 2

Note:
More support problems

EXAM NO.	ASME CAT	ASME ITEM NO.	IDENTIFICATION TAG NUMBER SUFFIX	CODE	DESCRIPTION	SYST	NDE METH	UNACCEPTABLE INDICATION AND DISPOSITION
000001374	B-K-1	B10.10	CF-0052WA	WELD	INTEGRAL ATTACHMENT WELD	CF	PT	INDICATION REMAINS UNCHANGED FROM PRIOR DATA.
000007175	F-A	F1.10	DH-0004	SUPPORT	DH REMOVAL SUPPORT	DH	VIS	SPALLED CONCRETE ADJACENT BASEPLATE ACCEPTABLE PER EVALUATION. REPLACED MISSING NUT ON PIPE CLAMP.
000007639	F-A	F1.10	MUE-0039	SUPPORT	MAKEUP HYDRAULIC SNUBBER	MU	VIS	BASE PLATE GAP DUE TO WALL CONTOUR EVALUATED AS ACCEPTABLE.
000007557	F-A	F1.10	MU-0139-3	SUPPORT	MAKEUP SUPPORT	MU	VIS	LOOSE LOCK NUT TIGHTENED.
000007247	F-A	F1.20	DHH-0187	SUPPORT	DH REMOVAL SUPPORT	DH	VIS	BOUND SNUBBER PADDLE EVALUATED AS ACCEPTABLE. NO ADJUSTMENT REQUIRED DUE TO PIPE MOVEMENT PREDICTIONS.
000008699	F-A	F1.20	DHH-0188	SUPPORT	DH REMOVAL SUPPORT	DH	VIS	BASE PLATE GAP DUE TO WALL CONTOUR EVALUATED AS ACCEPTABLE.
000007251	F-A	F1.20	DHH-0197	SUPPORT	DH REMOVAL SUPPORT	DH	VIS	BASE PLATE GAP EVALUATED AS ACCEPTABLE. BOUND SNUBBER PADDLE EVALUATED AS ACCEPTABLE BUT WAS ADJUSTED.
000007319	F-A	F1.20	EF-0116	SUPPORT	EMERGENCY FEEDWATER EF-U-0116	EF	VIS	TIGHTENED NUT ON PIPE CLAMP.
000007842	F-A	F1.20	EF-0163	SUPPORT	EMERGENCY FW SUPPORT	EF	VIS	BASE PLATE GAP EVALUATED AS ACCEPTABLE.
000008293	F-A	F1.20	MS-0193	SUPPORT	MAIN STEAM SUPPORT	MS	VIS	GAP BETWEEN SUPPORTING MEMBER AND WASHER/NUTS GREATER THAN SPECIFIED ON DRAWING. EVALUATION OF SUPPORT WAS ACCEPTABLE, ANY GAP EXCEEDING .25" WAS ADJUSTED.
000008294	F-A	F1.20	MS-0232	SUPPORT	MAIN STEAM SUPPORT	MS	VIS	CONCRETE SPALLING ON SIDES OF BASE PLATE AND BASE PLATE GAP REMAIN UNCHANGED FROM PREVIOUS EXAMINATIONS.

C-254

NIS-1 OWNERS DATA REPORT FOR INSERVICE INSPECTION

DATE: 02/03/92

PAGE: 0002

1. OWNER: Metropolitan Edison Company, 2800 Pottsville Pike,
Reading, Pennsylvania
2. PLANT: Three Mile Island Nuclear Generating Station
Route 441 South, Middletown, Pennsylvania
3. PLANT UNIT: 1
4. OWNER CERTIFICATION OF AUTHORIZATION (if required): None
5. COMMERCIAL SERVICE DATE: 09/02/74
6. NATIONAL BOARD NUMBER FOR UNIT: Reactor Vessel N-109

OUTAGE 9R TABLE 2

INTERVAL 2

EXAM NO.	ASME CAT	ASME ITEM NO.	IDENTIFICATION		DESCRIPTION	SYST	NDE METH	UNACCEPTABLE INDICATION AND DISPOSITION
			TAG NUMBER SUFFIX	CODE				
000008704	F-A	F1.20	MS-0235	SUPPORT	MAIN STEAM SUPPORT MS-U-0235	MS	VIS	BASE PLATE GAP DUE TO WALL CONTOUR EVALUATED AS ACCEPTABLE.
000008295	F-A	F1.20	MS-0308	SUPPORT	MAIN STEAM SUPPORT	MS	VIS	MINOR ANCHOR BOLT SLIPPAGE DUE TO THE SUPPORT BEING MISSADJUSTED. REPAIR BY ADJUSTING THE ROD AND RETORQUING THE ANCHORS.

Appendix D

Inquiry on Overlap of ISI Intervals

January 13, 1993

Secretary
ASME Boiler and Pressure Vessel Committee
345 East 47th Street
New York, NY 10017

SECTION XI IWA-2400 INQUIRIES - JFC-04-93

Reference: Section XI, IWA-2400, 1974 Edition through 1992 Edition

Dear Mr. Secretary:

Consideration of the following inquiries, to clarify Section XI inspection interval requirements, is requested.

Inquiry 1 - Is it the intent of Section XI that IWA-2400, which permits flexibility in the length of an inspection interval, permits overlap of inspection intervals, i.e. a subsequent interval may be started before the existing inspection interval is over?

Proposed 1 reply - No.

Inquiry 2 - Is it the intent of Section XI that, in the case of a new inspection interval starting during a plant outage, examinations for the new interval may be performed and credited during the same plant outage, i.e. perform inservice examinations for the new interval prior to the plant operating during the new interval?

Proposed 2 reply - No, it is the intent of Section XI that examinations in the new interval be performed following plant operation in the new interval.

Sincerely,

J. F. Cook, Sr.
Principal Investigator
Nondestructive Materials Characterization

cjs

cc: O. F. Hedden
T. J. Mawson
R. J. Tamminga

S. D. Weinman
T. M. Zwick

BIBLIOGRAPHIC DATA SHEET

(See instructions on the back of this form.)

TITLE AND SUBTITLE

Review of Section XI Inservice Inspection
Program Effectiveness

AUTHOR(S)

J. F. Cook, Sr.

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(Assigned by NRC. Add Vol., Supp., Rev.,
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EGG-MS-10781

2. DATE REPORT PREPARED

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Nondestructive Materials Characterization
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415-2209

PERFORMING ORGANIZATION: NAME AND ADDRESS (If NRC, provide Division Office or Region; U.S. Nuclear Regulatory Commission; and mailing address; if contractor, provide NRC Division Office or Region; U.S. Nuclear Regulatory Commission; and mailing address.)

Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUPPLEMENTARY NOTES

ABSTRACT (See instructions)

To evaluate the effectiveness of Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, searches were performed of the Licensing Event Report and Nuclear Plant Reliability Data System computerized data bases, and a review was made of inservice inspection summary reports. It was found that the Section XI examinations and tests detect flaws in welds and plant components and result in subsequent corrective action. This study also shows that the format and topics of information provided in Section XI-prescribed inservice inspection summary reports vary widely.

KEY WORDS DESCRIPTORS (Use words or phrases that will assist reviewers in locating the report.)

1. SUBJECT AREA

2. IDENTIFYING NUMBER

3. TITLE

4. AUTHOR

5. DISTRIBUTION STATEMENT

6. REPORT

END

**DATE
FILMED**

12/23/93