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In-Service Inspection Report of the ASME Boiler & Pressure Vessel Code..........

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Mr. Norman C. Moseley, Director
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DOCKETED USNRC AUG 6 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Dear Mr. Moseley:

H. B. ROBINSON UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
IN-SERVICE INSPECTION REPORT

As required by Technical Specification 6.9.1.d and in accordance with Section XI, IS-600, of the ASME Boiler and Pressure Vessel Code, 1970 Edition, the following in-service inspection report spanning the first five (5) years of commercial operation is submitted for your information.

The requirements of IS-622.3 require specific information to be provided in our report. The required data, as practical, is provided on Attachment A. The remainder of the required data is contained in the report, with the exception of information regarding the inspectors which is not pertinent to this summary. All data sheets, inspector and equipment documentation is at the facility available for inspection as you deem necessary.

The requirements for the in-service structural surveillance of the reactor vessel and primary system boundary of the H. B. Robinson Unit No. 2 Plant are defined in Section 4.2 of the Technical Specifications. Specific requirements as to which, and to what extent, components or areas of components are to be examined during the initial five years of plant commercial operation are detailed in Table 4.2.1.

Three refueling outages (March, 1973, May/June, 1974, and November, 1975) have occurred during the initial five years of plant operation, during which time In-service Inspection Programs have been conducted to comply with the requirements of the plant Technical Specification. A tabulation of the areas and extent of examination required by Table 4.2.1 of the Technical Specification is given in Attachment B.

7988

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All of the required examinations have been completed to the extent permitted by the limitations of the geometric configuration of welds and the access available in the H. B. Robinson Unit 2 Plant. The components and systems of the H. B. Robinson Plant were designed and fabricated before any regulatory requirements for the performance of in-service inspections were published and, as such, certain areas exist where the configuration of components limits the extent to which nondestructive examinations can be performed. This is specifically true of circumferential butt welds in piping systems at locations where the piping connects to fittings such as elbows, tees or reducers and valves or branch nozzles. In most such instances, ultrasonic examinations have been conducted from the pipe side of the weld only, with little or no examination performed from the surface of the fitting.

Two items specifically identified by Table 4.2.1 as being subject to the requirements of the In-service Inspection Program have not been examined in accordance with the requirements of Section XI of the ASME Code. These are:

1. Integrally welded piping system supports (IS-261 Item 4.5 Category K-1) are required to be both visually and volumetrically examined. All of the pipe support integrally welded attachments are by fillet weld. The configuration of these welds is such that meaningful results cannot be obtained by ultrasonic examination and surface examination of these welds has been performed in lieu of volumetric examinations. Authorization to substitute surface for volumetric examination techniques has been provided by Change Number 39 to the plant Technical Specifications.

2. Table 4.2.1 identifies valve supports and hangers (IS-261 Item 6.7 Category K-2) as requiring examination under the requirements of the In-service Inspection Program. There are no instances of valve supports or hangers occurring within H. B. Robinson Plant reactor coolant system pressure boundary and consequently no examinations have been performed.

The In-service Inspection Program for H. B. Robinson Plant is where practical, in compliance with Section XI of the 1970 Edition of the ASME Code as specified in the Technical Specifications. To provide further assurance of safety to the general public and safe reliable operation of the plant, Carolina Power & Light Company has, to the extent practical, extended the In-service Inspection Program performed on components and systems at the H. B. Robinson Plant beyond the requirements of the Technical Specifications in an attempt to fully comply with the requirements of the 1971 Edition of Section XI of the ASME Boiler and Pressure Vessel Code. To this end, in addition of the examinations required by Table 4.2.1 the following components and component areas have been examined.

IS-261 Reference	Area Examined	Method
1.2	Reactor Vessel Closure Head Meridional Welds	Volumetric
1.11	Reactor Vessel Conoseal Bolting	Visual
2.1	Pressurizer Circumferential Shell Welds	Visual and Volumetric
2.1	Pressurizer Longitudinal Shell Welds	Visual and Volumetric
2.7	Pressurizer Integrally Welded Supports	Visual and Volumetric
--	Pressurizer Nozzle to Safe-end Welds	Visual, Volumetric & Surface
3.1	Steam Generator (Loops A, B & C) Channel Head to Tube Sheet Welds	Visual and Volumetric
3.3	Steam Generator Nozzle to Safe-end Welds	Visual, Surface & Volumetric
3.1	Regenerative Heat Exchanger Head to Shell Welds	Visual and Volumetric
3.2	Regenerative Heat Exchanger Nozzle to Shell Welds	Visual and Surface
4.1	Reactor Coolant Pipe to Safe-end Welds	Visual, Surface and Volumetric
4.2	Piping System Nozzle Branch Connections	Visual and Volumetric
4.8	Piping System Socket Welds	Visual and Surface
5.5	Reactor Coolant Pump Seal Housing Bolting	Visual

A complete listing of examinations performed to meet the requirements of Section XI of the Code for the initial 40-month period is given in Attachment C. Additionally, surface examinations were performed on reactor vessel primary nozzle to safe-end welds to the extent permitted within the limited access provided in the sandbox opening and on the closure head cladding when the head was in the laydown area. Exception had been taken to the performance of these examinations in the Technical Specification due to anticipated radiation levels.

The regenerative heat exchanger in the H. B. Robinson Unit No. 2 Plant is not a Class A heat exchanger; however, due to lack of better guidance, IS-261, Item 3.1 and 3.2 have been utilized to classify areas of this vessel requiring examination. Rather than examine 1 1/2 percent of the total length of each shell weld, 100 percent of one shell weld was examined.

Section XI of the Code recognizes in IS-261 Items 4.7 and 4.8 that meaningful results cannot be obtained from the ultrasonic examination of branch nozzle connections of 4-inch nominal pipe size and smaller. Surface examinations were performed on the regenerative heat exchanger nozzle to shell welds rather than the volumetric examination which would be required by IS-261 Item 3.2.

Item 5.2 (Category L-2) of Technical Specification Table 4.2.1 requires a visual examination of the accessible pressure boundary surfaces of the reactor coolant pump casings if the pump internals are removed. During the May, 1975 repair of "C" reactor coolant pump, the accessible pressure boundary surfaces were visually inspected to meet this requirement.

The first and second reactor vessel material irradiation surveillance capsules were removed and analyzed as required by Technical Specification Table 4.2.1 Item 7.2. The report on the analysis of the first capsule was submitted to the Directorate of Licensing by Carolina Power & Light Company in our letter of December 20, 1973. The second capsule was removed during the November, 1975 refueling outage and is presently being tested and analyzed. Upon completion of this analysis, Carolina Power & Light Company will provide the Nuclear Regulatory Commission with the results of this analysis.

Item 1.15 (Category N) of Technical Specifications Table 4.2.1 requires a visual examination of interior surfaces and internal components of the reactor vessel (as accessible) to be performed during the first refueling outage. This examination was performed, as required, and reported to the Nuclear Regulatory Commission in the H. B. Robinson Unit No. 2 Semi-Annual Operating Report No. 6.

Examinations performed during the initial plant outage in 1973 revealed an indication in the Loop A Steam Generator Channel Head to tube sheet weld. This same indication was monitored during the 1974 outage and reported as having no change in size. Again during the 1975 outage the indication which had previously been reported as being present in the Loop A Steam Generator Channel Head to tube sheet weld was re-examined to confirm that no change in size had occurred. The indication was again recorded as being present for a length of 22 inches in the same location but at a considerably reduced amplitude of 35 percent of the reference calibration level. The apparent discrepancy in recorded amplitudes is easily explained by a comparison of the procedures utilized. The examination performed during this recent outage was conducted in accordance with Procedure ISI-5 Rev. 12 which prohibits (in accordance with current Code practice) the performance of a transfer.

During the performance of the previous examinations a transfer had been performed wherein the back surface response of the calibration block was compared with the back surface response from the channel head casting and the instrument gain increased to obtain equivalent values. The back surface of the channel head is not parallel to the front and is covered with clad in a rough "as welded" condition. This condition obviously results in a situation where a signal reflected from the back surface will always be considerably weaker than that from the parallel back surface of the calibration block. Increasing instrument gain to obtain an equivalent signal results in the examination being performed at an abnormally high sensitivity which greatly exaggerates the response from any included reflector.

The currently utilized examination technique being performed at calibration sensitivity results in a more realistic recording level of any reflector present in the material. Consequently, the reflector present in the Loop A Channel Head to tube sheet weld is considerably smaller (in amplitude) than previously determined and it seems probable that further monitoring of this reflector should not be required.

An evaluation of the reflector in "A" steam generator was made which determined the indication to be slag which is within the requirements of the ASME Code, and does not affect the safe operation of the plant.

One relevant liquid penetrant indication was recorded on the Pressurizer Spray Line, Weld #1. This indication was an approximate 1/2 inch linear indication in the elbow to safe-end weld and was due to an unground weld fold area. This indication was filed out and evenly blended in with the surrounding base metal, then reinspected and found to be free of indications.

There were no relevant reflectors observed by ultrasonic examination during the May-June 1974 refueling outage. The "A" reactor coolant pump flywheel gage hole plugs could not be removed thus limiting ultrasonic examination of that item to a periphery scan function. Ultrasonic examination commitments were extended to include the upper support bolts on all reactor coolant pumps (18 per pump, all exceeding 2 inches in diameter). Ultrasonic examination was used to supplement the visual and impact testing program on the reamining support bolting in the steam generators and reactor coolant pumps. In particular, steam generator lower support studs were examined ultrasonically to verify no fracture existed when an impact (hammer) test reflected a possible problem on four (4) studs. The four (4) studs were found to be acceptable and the impact test results were determined to be caused by a lack of nut tightness.

Surface (liquid penetrant) examination results were acceptable with no relevant indications observed in 1974. Visual examinations during this refueling outage were a major effort of the inspection program. Class I component support bolting was examined with the following items reported.

a. The "B" steam generator upper support pad (between the hot leg manway and hot leg nozzle) outboard corner bolt was found missing and subsequently replaced at a later date.

b. Impact (hammer) testing of four (4) lower steam generator support studs revealed a possible problem which was resolved by ultrasonic examination as discussed above.

The bolting impact test sample was increased from 25 percent to 100 percent of all lower support bolting because of significant results described for the four studs discussed previously. This impact testing was proven to be a highly reliable examination method.

A complete baseline (100 percent) examination (visual) was performed of all Class I piping systems' hangers and supports for systems within the boundaries required, to permit the establishment of an Inservice Inspection Program for piping supports and hangers. This baseline examination which provides the basis for establishing the 10-year piping supports and hangers plan is credited to the first 3 1/3-year interval. Several loose rods and U-Bolts were found and corrected during this examination.

The November, 1975 refueling outage in-service inspection completed the inspection requirements for the five-year interval in accordance with the H. B. Robinson Technical Specifications. Examinations during this outage consisted of visual, surface and volumetric examinations of various components. The inspection results and the corrective action taken (except for minor hanger corrections) are as follows:

1. <u>Pressurizer Manway Bolts</u>

   Indication: Minor rust on four bolts.
   Corrective Action: The bolt surfaces were cleaned.

2. Reactor Vessel Studs, Nuts and Washers

Indication: Nicked threads on Studs #17, 18 and 20.
Corrective Action: The three studs were "dressed" and the affected areas were reinspected prior to reuse.

3. Reactor Vessel Conoseal Bolting

Indication: One bolt on "A" Conoseal had galled threads.
Corrective Action: A new bolt was installed.

4. Steam Generator Primary Manway Bolting

Indication: Various bolts had nicked or eroded threads, and required cleaning.
Corrective Action: All primary manway bolts were brushed, thread nicks were "dressed" and two bolts were replaced.

5. Loop "B" 14-Inch RHR Suction Pipe

Indication: Linear indications were noted on support lug welds during the liquid penetrant inspection.
Corrective Action: All weld defects were ground out, repaired, and the finished welds were subsequently reinspected satisfactorily.

In addition to the automatic "bug and track" ultrasonic examination of loops "A" and "B" Hot Leg Reactor Vessel to safe-ends, a surface (liquid penetrant) examination was performed on approximately 25 percent of each weld (the accessible area) and a 100 percent visual examination of each weld was performed using a fiberscope which was mounted on the "bug" and traversed the full length of each weld. No indications were detected during these 1975 examinations.

As all examinations to satisfy the requirements of the Technical Specifications for the first five (5) years of plant operation have been completed, Carolina Power & Light Company intends to perform additional

examinations to satisfy the requirements of Section XI of the Code, to the extent practical, for the second 40-month period. Carolina Power & Light Company will also continue to, as feasible and allowed by geometric configurations, expand the H. B. Robinson Plant In-Service Inspection Program to meet the requirements of the current Code requirements.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

CSB:bb

Attachments

H. B. ROBINSON UNIT NO. 2
INSERVICE INSPECTION REPORT

ATTACHMENT A

Date:	May 19, 1976
Facility Owner and Address:	Carolina Power and Light Company 336 Fayetteville Street P. O. Box 1551 Raleigh, North Carolina 27602
Facility Name and Address:	H. B. Robinson Plant Carolina Power & Light Company P. O. Box 790 Hartsville, South Carolina 29550
Name Assigned To Nuclear Power Plant:	H. B. Robinson Unit No. 2
Commercial Service Date:	March 7, 1971
Gross Generating Capability:	739 MW(e)
Dates of Inservice Inspections Performed:	March 19 - 28, 1973 May 21, 1974 - June 13, 1974 November 3 - 17, 1975

MAJOR COMPONENT DATA

Component	National Board Number Assigned	Additional Information	Manufacturer
Reactor Vessel	20772	Vessel Serial No. 66109 Vessel Head Serial No. 66209 Year Manufactured 1968	Combustion Engineering Inc. 911 West Main Street Chattanooga, Tennessee
Pressurizer	722	Vessel Serial No. B4777 Year Manufactured 1968 Contract No. 9-8799	Chicago Bridge & Iron Company Birmingham, Alabama
Steam Generator "A"	750	Vessel Serial No. 16-A-6081-1 Year Manufactured 1968	Westinghouse Electric Corp. Heat Transfer Division P. O. Box 9175 Lester, Pennsylvania 19113
Steam Generator "B"	754	Vessel Serial No. 16-A-6081-3 Year Manufactured 1969	Westinghouse Electric Corp. Heat Transfer Division P. O. Box 9175 Lester, Pennsylvania 19113
Steam Generator "C"	752	Vessel Serial No. 16-A-6081-2 Year Manufactured 1969	Westinghouse Electric Corp. Heat Transfer Division P. O. Box 9175 Lester, Pennsylvania 19113

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF TECH. SPEC. FOR FIRST 5 YEARS
REFUELING OUTAGES: 1, 2, AND 3

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION	TYPE OF EXAMINATION
Reactor Vessel	1.3	Closure Head to Flange Weld	33%	Volumetric
	1.3	Vessel to Flange Weld	33%	Volumetric
	1.7	Primary Nozzle to Safe End Welds	33%	Volumetric
	1.8	Closure Studs and Nuts	33%	Volumetric & Visual
	1.9	Ligaments between threaded stud holes	33%	Volumetric
	1.10	Closure Washers	33%	Visual
	1.13	Closure Head Cladding	3 Patches	Visual and Surface
	1.15	Vessel Internals	First R/F outage	Visual
Pressurizer	2.6	Manway Bolts	33%	Visual
Stm. Gen. Loop A	3.5	Manway Bolts	33%	Visual
Stm. Gen. Loop B	3.5	Manway Bolts	33%	Visual
Stm. Gen. Loop C	3.5	Manway Bolts	33%	Visual
Reactor Coolant Piping	4.2	Circumferential Butt Welds	5%	Volumetric & Visual
Associated Auxiliary Piping	4.2	Circumferential Butt Welds	5%	Volumetric & Visual
	4.4	Pressure Retaining Bolting	33%	Visual
	4.5	Integrally Welded Supports	33%	Visual and Surface or Volumetric
	4.6	Pipe Supports and Hangers	33%	Visual
Reactor Coolant Pump (Loops A, B, & C)	5.4	Main Flange Bolting	33%	Volumetric & Visual
	5.7	Support Structures	33%	Visual

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF TECH. SPEC. FOR FIRST 5 YEARS
REFUELING OUTAGES: 1, 2, AND 3

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION	TYPE OF EXAMINATION
Valves	6.5	Valve Bonnet Bolting	33%	Visual
	6.7	Supports and Hangers	33% Not Applicable	
Reactor Coolant Pump (Loops A, B, & C)	7.1	Flywheels	Visual at 1st RF	
Reactor Vessel	7.2	Reactor Vessel Material Irradiation Surveillance Specimen	Capsule 1 and 2	Tensile and Charpy V Notch

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF IS-261 ASME CODE, SECTION XI, 1971 EDITION FOR FIRST 3 1/3 YEAR PERIOD

REFUELING OUTAGES: 1 AND 2

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION
Reactor Vessel	1.1	Longitudinal and circumferential shell welds in core regions (7)	Scheduled outage 8
	1.2	Closure head meridional welds (6)	2.5% of each weld (2")
	1.3	Closure head to flange weld	25% (117 inches)
	1.3	Vessel to flange weld	Scheduled outages 3, 4, 5, and 8
	1.4	Nozzle to shell welds (6)	Scheduled outage 8
	1.5	Vessel penetrations including control rod drive penetrations and control rod housing pressure boundary welds.	Not applicable
	1.6	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.	Done during hydrostatic at or near end of outage 8
	1.7	Primary nozzle to safe end welds	33% (2 nozzles)
	1.8	Closure studs and nuts	33% (17 studs and nuts)
	1.9	Vessel flange ligaments	Scheduled outages 3, 4 & 5 and outage 8.
	1.10	Closure washers	33% (17 washers)
	1.11	Conoseal bolting	33% (5 bolts)
	1.12	Integrally welded vessel supports	Scheduled outage 8
	1.13	Closure head cladding	2 patches
	1.14	Vessel cladding (6 patches)	Scheduled outage 8.
	1.15	Vessel internals	At first refueling outage.

Attachment C

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF IS-261 ASME CODE, SECTION XI, 1971 EDITION FOR FIRST
3 1/3 YEAR PERIOD

REFUELING OUTAGES: 1 AND 2

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION
Pressurizer	2.1	Circumferential shell welds	1.5% (4.5 inches)
	2.1	Longitudinal shell welds	3% (3 inches)
	2.2	Nozzle to vessel welds	Done during hydrostatic test
	2.3	Heater connections	Done during hydrostatic test
	2.4	Heater connections and instrument and sample nozzles	Done during hydrostatic test
	2.5	Pressure retaining bolting	Not applicable
	2.6	Manway bolts	25% (4 bolts)
		Nozzle to safe end welds	2 Nozzles
	2.7	Integrally welded vessel supports	3% (9 inches)
	2.8	Vessel cladding (1 patch)	Scheduled outage 8
Steam Generator Loop A	3.1	Channel head to tube sheet weld	1.5% (6 inches)
	3.2	Nozzle to shell welds	Not applicable
	3.3	Nozzle to safe end welds	2 nozzles
	3.4	Pressure retaining bolting	Not applicable
	3.5	Manway bolts	25% (8 bolts)
	3.6	Integrally welded supports	Not applicable
	3.7	Vessel cladding (1 patch)	Scheduled outage 8
Steam Generator Loop B	3.1	Channel head to tubesheet weld	1.5% (6 inches)
	3.2	Nozzle to shell welds	Not applicable
	3.3	Nozzle to safe end weld	Scheduled outages 3,4 & 5 & 6, 7 & 8
	3.4	Pressure retaining bolting	Not applicable
	3.5	Manway bolts	25% (8 bolts)

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF IS-261 ASME CODE, SECTION XI, 1971 EDITION FOR FIRST 3 1/3 YEAR PERIOD

REFUELING OUTAGES: 1 AND 2

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION
Stm. Gen'r Loop B (Contd)	3.6	Integrally welded supports	Not applicable
	3.7	Vessel cladding (1 patch)	Scheduled outage 8
Steam Generator Loop C	3.1	Channel head to tubesheet weld	1.5% (6 inches)
	3.2	Nozzle to shell welds	Not applicable
	3.3	Nozzle to safe end weld	2 nozzles
	3.4	Pressure retaining bolting	Not applicable
	3.5	Manway bolts	25% (8 bolts)
	3.6	Integrally welded supports	Not applicable
	3.7	Vessel cladding (1 Patch)	Scheduled outage 8
Regenerative Heat Exchanger	3.1	Head to shell welds	2 welds
	3.2	Nozzle to shell welds	1 weld
	3.3	Nozzle to safe end welds	Not applicable
	3.4	Pressure retaining bolting	Not applicable
	3.5	Manway bolts	Not applicable
	3.6	Integrally welded supports	Not applicable
	3.7	Vessel cladding	Not applicable
Reactor Coolant Piping	4.1	Pipe to safe-end welds	6 welds (included) in items 1.7 and 3.3)
	4.2	Circumferential butt welds	10% (3 welds)
Associated Auxiliary Piping	4.1	Pipe to safe end welds	2 welds
	4.2	Circumferential butt welds	10% (30 welds)
	4.2	Nozzle root connections	20% (1 weld)
	4.3	Pressure retaining bolting	Not applicable

SCHEDULE OF INSPECTIONS

TO MEET COMMITMENTS OF IS-261 ASME CODE, SECTION XI, 1971 EDITION FOR FIRST 3 1/3 YEAR PERIOD

REFUELING OUTAGES: 1 AND 2

COMPONENTS	IS-261 REFERENCE	AREA TO BE EXAMINED	EXTENT OF EXAMINATION
Associated Aux. Piping (Cont'd)	4.4	Pressure retaining bolting	32% (48 bolts)
	4.5	Integrally welded supports	100% (visual only)
	4.6	Piping supports and hangers	100% (253 supports)
	4.7	Circumferential and longitudinal pipe welds and branch pipe connections	Not applicable
	4.8	Socket welds	10% (41 welds)
Reactor Coolant Pump Loops A, B and C	5.1	Pump casing welds	Scheduled outage 8
	5.2	Pump casing internal surface	Scheduled outage 8
	5.3	Nozzle to safe end welds	Not applicable
	5.4	Flange bolting	33% (24 bolts)
	5.5	Seal housing bolting	33% (18 bolts)
	5.6	Integrally welded supports	Scheduled outage 8
	5.7	Support structures	33%
Valves	6.1	Valve body welds	Not applicable
	6.2	Internal surfaces of valve bodies	Scheduled outage 8
	6.3	Valve to safe end welds	Not applicable
	6.4	Pressure retaining bolting	Not applicable
	6.5	Valve bonnet bolting	100% (382 bolts)
	6.6	Integrally welded supports	Not applicable
	6.7	Supports and hangers	Not applicable
Reactor Coolant Pump (Loops A, B & C)	7.1	Flywheels	Visual without disassembly
	7.2	Irradiation specimen schedule	Capsule 1 at First refueling