U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-318/87-10

Docket No. 50-318

License No. DPR-69

Licensee: Baltimore Gas and Electric Company

Facility Name: Calvert Cliffs Nuclear Power Station

Inspection At: Lusby, Maryland

Inspection Conducted: April 6 through 10, 1987

Inspector: A J Lodewyk — 6/3/87
A. J. Lodewyk, Reactor Engineer — date

Approved by: Jack Strosnider — 6/3/87
J. R. Strosnider, Chief, Materials and Processes Section, EB, DRS — date

Inspection Summary: A routine, unannounced inspection was conducted by one region based inspector of activities associated with the licensee's Inservice Inspection Program and Steam Generator integrity. Those areas examined included review of ISI procedures and data records, observation of nondestructive examination activities, and actions pertaining to steam generator surveillance and maintenance.

Results: During this inspection, no violations were identified.

DETAILS

1.0 Persons Contacted

Throughout this inspection, various site and corporate office representatives were interviewed for information and auditing purposes. Those persons contacted include the following:

Baltimore Gas and Electric Company

* R. Allen, Principal Engineer, Performance
* S. Cowne, Senior Engineer, Licensing
* P. T. Crinigan, General Supervisor, Chemistry
L. Decker, Engineer, Performance
* W. J. Lippold, Manager Nuclear Engineering Services
J. Macklin, Engineer, Performance
* B. Rudell, Senior Engineer, Performance

Factory Mutual

* R. Lawrence, ANII

U. S. Nuclear Regulatory Commission

T. Foley, Senior Resident Inspector
* D. Trimble, Resident Inspector

* Denotes those persons present at the exit meeting conducted at the close of this inspection on April 10, 1987.

2.0 Inservice Inspection Activities

During the current refueling outage, the Calvert Cliffs, Unit 2, Inservice Inspection Program included volumetric examination of the reactor pressure vessel. The vessel was designed and constructed by Combustion Engineering in accordance with the 1965 Edition of the ASME Boiler and Pressure Vessel Code. Vessel construction included welding of 9 inch thick carbon steel plates. The vessel has a 1/4 inch thick weld deposited stainless steel clad on the inner surface. The required Inservice Inspection volumetric examination was performed by Southwest Research Institute (SwRI) in accordance with ASME B&PV Code, Section XI, 1974 Edition, Summer 1975 Addenda. The equipment and techniques used to ultrasonically examine the reactor vessel included:

- PAR device transducer manipulator
- 0, 45 and 60 degree transducers calibrated on 3/16 inch holes (for ASME Code volumetric exam)
- 50/70 degree dual transducer calibrated on 1/16 inch near surface holes (for Regulatory Guide 1.150 near surface exam)
- strip chart and video screen recordings
- conventional (manual) signal interpretation

In addition to the standard vessel examination, BG&E completed a secondary analyses of the ultrasonic data signals using the digital analysis equipment: UDRPS (Ultrasonic Data Recording and Processing System). BG&E also elected to expand the scope of the 10-year reactor vessel inspection plan to include 100% of the accessible areas of the belt-line region welds.

During the standard (SwRI) ultrasonic examination, an indication was found in the reactor core belt-line region using the 50/70 degree dual transducer. The indication is located in longitudinal seam weld no. 2-203A, approximately 216 inches below the seal surface and in a plane at a slight angel to the inside surface of the vessel. The 50/70 signal data characterized the indication as near as 0.7 inches below the clad surface and covering an area approximately 0.8 x 1.0 inches with a maximum depth of 1.5 inches. The 0, 45 and 60 degree data as well as previous (baseline) ultrasonic examination data of this area did not record an indication. The difference in data signals is justifiable as the ASME Code and baseline ultrasonic examination techniques are not specifically designed to detect near surface flaws. The RG 1.150 (50/70 degree) examination is primarily intended to detect near surface flaws.

The SwRI, ASME Code evaluation determined the indication is acceptable as a subsurface flaw having no measurable (less than 1/16 inch) through-wall dimension. Flaw sizing was obtained using a standard amplitude based technique. SwRI analyses was completed in accordance with ASME Section XI, IWB 3500 and RG 1.150 requirements.

In conjunction with the standard evaluation of the ultrasonic data, UDRPS results were considered by BG&E to determine the acceptability of the vessel indication. UDRPS analyses of the original 40 and 60 degree data signals confirmed the indication within the weld at the same location. For sizing and further resolution of the indication using UDRPS, BG&E elected to re-scan the suspect area using two focussed dual element refracted L-wave transducers (45 and 60 degree). The re-scan was performed using 0.2 inch increments and the sizing technique was time of flight tip defraction method. The final examination results using UDRPS determined the indication to be two parallel indications with rounded ends (i.e. not crack-like). The largest indication was characterized as approximately 1.3 inches long, 0.2 inches in depth and 0.7 inches below the inner vessel surface. The second, parallel indication was analyzed to be 0.4 inches offset from the first indication, approximately 0.5 inches long, 1.5 inches from the inner surface and 0.2 inches in depth. The uncertainty in measurements is a result of the sizing technique used and the indications rounded end data characteristics.

Independent of the ultrasonic data evaluation of the indication(s), BG&E retrieved the vessel construction radiographs for the suspect area. Licensee and NRC review of the radiographs agreed that the ultrasonic data signals were being initiated from a construction defect. The radiographic film review confirmed the size and orientation of the evaluated ultrasonic indications corresponded to an apparent slag inclusion. The weld radiograph revealed that the larger of the two ultrasonic indications was in fact two end-to-end shorter inclusions. Although the indication was determined acceptable to ASME Code standards, BG&E representatives intend to complete an engineering fracture mechanics analyses to ensure the integrity of the Unit 2 reactor vessel, using the larger, combined indication length to be conservative. Based upon comparison of the final ultrasonic data analyses and the radiographic indications, it appears the weld inclusion has not increased during service.

The inspector observed portions of the data acquisition, analyses and evaluation throughout the reactor vessel inspection. Based upon these observations, discussions with cognizant licensee personnel, data records and radiographic film reviews, the inspector determined the licensee completed a thorough, technical review of the suspect area. The inspector had no further questions regarding this matter at this time.

3.0 Licensee Activities to Maintain Steam Generator Integrity

The licensee's steam generator (S/G) preventive maintenance program (as it is related to maintaining S/G integrity and controlling station secondary water chemistry) was reviewed relative to criteria, commitments, and recommendations provided in:

-- Condition II.C.7 of License No. DPR-69 regarding a secondary water chemistry monitoring program.

-- Electric Power Research Institute (EPRI) Report NP-2704-SR, "PWR Secondary Water Chemistry Guidelines", Revision 1 (1984).

-- NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity".

-- Baltimore Gas & Electric response to Generic Letter 85-02; "Staff Recommended Actions Stemming From NRC Integrated Program For The Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity

Licensee performance in S/G preventive maintenance was determined by interviews with chemistry, radiation protection, maintenance, quality assurance, and plant engineering personnel; review of selected procedures, reports and records; and observations of plant facilities and equipment during facility tours. The review addressed the licensee's maintenance, Inservice Inspection (ISI) and ALARA program procedures, policies and implementation.

Those plant procedures, records and reports selected for review were found to contain:

- critical chemical variables and limit/action levels for secondary water chemistry control

- monitoring and inspection of condenser internals (including periodic Eddy Current inspection of condenser tubes when practicable)

- cleanliness, personnel and material accountability requirements

- ISI Program full length tube inspections and expanded Eddy Current testing samples

- use of equipment to minimize personnel radiation exposures (e.g. full S/G mock-up training and zero-entry SM-10 Eddy Current probe manipulator)

The procedures reviewed were consistent with license conditions and commitments and generally followed the guidelines established by EPRI. Review of secondary water chemistry data for the past operating cycle determined normal chemical concentrations are maintained within administrative established guidelines. Administrative controls and guidelines have been established for significant deviations in key chemistry parameters.

During this outage, the steam generator Inservice Inspection sample size exceeded the Technical Specification minimum requirements. Also, the licensee has scheduled a secondary analyses of eddy current data to be performed prior to start-up of the unit S/G's. Steam generator tube degradation has been experienced primarily in areas located just above the tube sheet(s). Over the past five years, the number of degraded and defective steam generator tubes has increased. The licensee has expanded the eddy current inspection scope as required by the license technical specifications. Recent, 100% inspection of steam generator tubes and use of upgraded eddy current inspection equipment assures the detection of service induced flaws.

At the time of this inspection, the results of previous, Unit 1 S/G tube pulls had not been completed to quantify the type(s) of tube degradation found at Calvert Cliffs Nuclear Power Station. Upon completion, both the results of the tube analyses and the results of the Unit 2 Eddy Current testing will be subject to further NRC review.

Overall, the licensee's preventive maintenance and inspection activities to maintain S/G integrity appear to meet industry recommendations and technical requirements. The inspector had no further questions at this time.

4.0 Exit Meeting

The inspector met with the licensee representatives (denoted in paragraph 1) at the close of this inspection on April 10, 1987. The inspector summarized the purpose, scope, findings and observations during this inspection. At no time during this inspection was written material provided to the licensee by the inspector.