UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-259/92-38, 50-260/92-38, and 50-296/92-38

Licensee: Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-259, 50-260, and 50-296

License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry 1, 2, and 3

Inspection Conducted: November 2-6, 1992

Inspectors: for J. L. Coley Jr. — 11/18/92 Date Signed

for R. C. Chou — 11/18/92 Date Signed

Approved By: J. J. Blake, Chief — 11/18/92 Date Signed
Materials and Processes Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted on site in the areas of inservice inspection (ISI) - observation work and work activities; review of radiographic film for class 1 reactor water clean up (RWCU) welds; observation of Unit 1, shroud manway access hole cover, ultrasonic (UT) examinations; and review of UT data (Information Notice [IN] 88-03 and IN 92-57 "Cracks in Shroud Support Access Hole Cover Welds"); review of TVA actions with regards to IN 92-35 "Higher Than Predicted Erosion/Corrosion in Unisolable Reactor Coolant Piping Inside Containment"; review of TVAs purposed response to NRC Generic Letter (GL) 88-01 Supplement 1 "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) In BWR Austenitic Stainless Steel Piping"; review of previously open NRC items; and review of pipe support design calculations for Unit 2.

Results:

One cited Violation No. 50-260,296/92-38-01," Inadequate Design Controls for Pipe Support Calculations," (paragraph 7); one non-cited violation No.50-260/92-38-02, "Failure to Properly Identify Support Spring Can Variability," (paragraph 7); and one unresolved item No. 50-296/92-38-03, "Evaluation of Weld Conditions," (paragraph 3) were identified by the inspectors. No deviations were identified. One weakness was also identified in the license's evaluation of information Notice No. 92-35. (paragraph 5) Not withstanding the items identified, the inspectors concluded that TVA management is actively involved in attempting to correct the root causes of problems with hardware and personnel performance. A balance of improved supervision, personnel training, employee commitment, and improved procedures appear to be obtaining positive results.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

R. Baird, Principle Civil Engineer
*O. Butler, Level III Examiner, Inspection Services
*R. Cutsinger, Lead Civil Engineer
*J. Davenport, Regulatory Engineer
*S. Fox, Level III Examiner, Inspection Services
*F. Froscello, ISI Supervisor
*E. Hartwig, Project Manager
T. Knuettel, Licensing Engineer
*L. Madison, Supervisor, Civil Engineering
*D. Massey, Regulatory Engineer
J. McCord, Stress Analyst
R. Phillips, Supervisor, Material Engineering
*D. Nye, Recovery Manager
G. Strickland, Material Engineering, Corporate Office
*J. Sabados, Chemistry and Environment Manager
*M. Turnbow, Manager, Inspection Services
*O. Zeringue, Site Vice President

Other license employees contacted during this inspection included engineers, mechanics, technicians, and administrative personnel.

Other Organizations

General Electric Nuclear Energy (GENE)

T. Brinkman, Project Manager, NDE Application Technology
M. Hart, Quality Assurance Manager
*R. Seals, ISI Supervisor,
S. Stanford, ISI Level II Examiner

NRC Resident Inspectors

*J. Munday, Resident Inspector
*C. Patterson, Senior Resident Inspector

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Inservice Inspection - Observation of Work and Work Activities Unit 3 (73753)

The licensee is currently in the process of preparing Unit 3 for restart. As part of this effort the licensee has instituted an integrated program to implement the commitments made in response to GL 88-01, NRC's position on IGSCC in boiling water reactors (BWRs) austenitic piping. GL 88-01 requires mitigation of IGSCC in susceptible piping by inspection, repair and or replacement. To comply with the requirements of GL-88-01, the piping runs exposed to fluid temperatures greater than 200 °F have been replaced with type 316 NG (Nuclear Grade) stainless steel, which is not susceptible to IGSCC. Piping runs not exposed to fluid temperatures greater than 200 °F are being replaced with type 316 stainless steel.

To date, the 12" dia. and 20" dia. portions of the reactor recirculation and the 6" dia. portions of the reactor water clean-up (RWCU) systems have been replaced inside containment. The 28" dia. recirculation piping that previously had reported IGSCC are in the process of having full structural design weld overlays applied.

Installation and welding of replacement piping was accomplished by General Electric (GE) Company, Nuclear Services and Project Department under the umbrella of the GE Quality Assurance (QA) Program. After each pipe weld is acceptable by radiography, mechanical stress improvement is performed. Nondestructive examinations (NDE), including preservice examinations, of the pipe welds are being performed by TVA.

The applicable Codes for the pipe replacement project are:

> American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME, B&PV) Code Sections III, V, and XI 1986 Edition.
>
> ASME, B&PV Code Section II and IX, latest edition.

The inspectors observed the preservice ultrasonic examination, performed on Weld No. RWCU-3-001-G023, to determine whether the approved NDE procedure (N-UT-18, Revision 13) was being followed; whether the examination personnel were knowledgeable of the examination method; and whether the examiner made the appropriate interpretation of the test results.

Within the areas examined, no violations or deviations were identified.

3. Review of Radiographic Film Unit 3 (57090)

The inspectors reviewed radiographic film, and associated records, for Class I, Reactor Water Clean-up (RWCU) Welds to determine whether the radiographs were prepared, evaluated, and maintained in accordance with the applicable Codes (ASME Sections III, and V, 1986 Edition).

Included in this review was verification that the penetrameters were the correct type, size, and properly placed on the pipe, and if adequate sensitivity was obtained with the radiographic technique. The radiographs were also reviewed to ensure that film density was within the allowable code variation, weld coverage was complete, and welding discontinuities were properly evaluated. Radiographs for the following welds were reviewed:

Weld Identification No.	Work Plan No.	Size
RWCU-3-001-G001	3250-92	6"Dia.
RWCU-3-001-G011	3250-92	6"Dia.
RWCU-3-001-G014	3539-92	6"Dia.
RWCU-3-001-G015	3539-92	6"Dia.
RWCU-3-001-G016	3539-92	6"Dia.
RWCU-3-001-G017	3539-92	6"Dia.
RWCU-3-001-G018	3539-92	6"Dia.
RWCU-3-001-G019	3539-92	6"Dia.
RWCU-3-001-G020	3539-92	6"Dia.
RWCU-3-001-G021	3539-92	6"Dia.
RWCU-3-001-G022	3539-92	6"Dia.
RWCU-3-001-G023	3539-92	6"Dia.
RWCU-3-001-G024	3634-92	6"Dia.
RWCU-3-001-G025	3634-92	6"Dia.
RWCU-3-001-G026	3634-92	6"Dia.

The inspector's review identified two weld radiographs which contained indications of welding conditions that the license should have investigated further.

The first weld was RWCU-3-001-G024 which had three film segments, (0-1, 2-3,and 3-0) where the consumable "K" insert ring had not been consumed. In each case the insert had melted to the point that there was no lack of fusion at either side of the weld prep but had not melted to the point that the constituents in the ring had flowed, (i.e. the insert ring still maintained its original shape.) The Code does not provide acceptance criteria for incomplete insert melt, so one option would be to apply the criteria for elongated indications. If this criteria was applied the weld would be rejectable; however, there appears to be a valid argument for not using this criteria since the ring did melt to the point of fusion and therefore the weld joint may have the structural soundness required.

Cognizant TVA management was notified and although the licensee felt the weld was acceptable they offered to have a metallurgical evaluation made of the weld soundness with this condition. In addition, TVA offered to research ASME Code Cases and Code Inquiries, as well as other industry guidance, in order to establish procedural guidelines which would give definitive acceptance criteria for this condition. TVA actions with regard to this weld are very responsible since this is a 316NG stainless steel weld which has received mechanical stress improvement. Any repair to this material would have some detrimental effects on its ability to mitigate IGSCC.

The second weld condition questioned by the inspectors was on Weld No. RWCU-3-003-G026. The Radiographic Inspection Report (RIR) for this weld had a comment adjacent to the evaluation for film segment 1-2 that a weld condition noted on the radiograph was a weld cap condition. Due to the magnitude (apparent depth) of the indication, the inspectors questioned whether there might also be a lack of fusion indication, on the side of the weld prep edge, along with the weld cap edge condition.

During the discussions with the licensee, the inspectors also learned that the examiners who had originally evaluated the radiograph had not visually examined the weld to confirm the weld cap condition. A visual examination was subsequently performed by the radiographic examiner and the inspectors; however, the weld reinforcement had been ground off for ISI, which would have removed any weld cap indication. Therefore, cognizant management personnel stated that another radiographic shot would be made of this film segment in order to assure that the questioned condition was not partly the result of lack of fusion.

The actions proposed by the license for the further evaluation of weld conditions noted by the inspectors should be adequate to resolve whether these conditions are acceptable or rejectable. This issue however, will be identified as unresolved item 50-296/92-38-03, "Evaluation of Welding Conditions", pending the results of the license's evaluations.

Within the areas examined, no violation or deviation was identified.

4. Observation of the Shroud Manway Access Hole Cover Examinations and Review of Ultrasonic Data, Unit-1 (92701)

NRC, IN 88-03 "Cracks in Shroud Support Access Hole Cover Welds", and IN 92-57 "Radial Cracking of Shroud Support Access Hole Cover Welds" alerted licensees of boiling water reactors (BWRs) of the potential for cracks in the welds of covers to the shroud support access holes within the reactor vessel. Each BWR has two access hole covers in the shroud plate, one at 0 degrees and the other at 180 degrees. The access hole covers for the Browns Ferry Unit 1 reactor were inspected by UT examination in late April, 1992. Circumferential cracking was detected, and a visual inspection confirmed that the cracks were through-wall. At that time the General Electric's (GE's) UT inspection fixture was not configured to detect radial cracks, but radial indications were detected in the 0° access hole cover during visual examination. As a result GE recommended that additional UT inspections be performed from the inside of the reactor with a modified UT fixture for radial scanning. In addition GE established methods to examine the reactor vessel and the shroud attachment welds inside the vessel from outside of the reactor vessel, in order to determine the extent of the radial cracking.

On November 2, 1992, the inspectors arrived at the Browns Ferry Nuclear Plant to observed the UT examinations of the shroud manway access cover plate welds and to review the recorded data. At this time GE had completed all scans from inside the vessel on the 0° manway access cover plate weld, but had not fully reviewed or plotted the data. Since GE was in the process of setting up the Smart 2000 to perform the UT examinations on the 180° manway cover from inside the vessel the inspectors started their review with the recorded 0° data. This review revealed that the scans from the ledge side and the cover side of the manway access cover had detected significant circumferential reflectors

indicative of IGSCC. However, the data from the scans to detect the radial indications was inconclusive, in that no apparent reflector was producing a signal indicative of IGSCC.

On November 3, 1992, the inspectors observed the UT examinations performed on the 180° manway access cover. During these examinations the inspector noted that the fixture used to detect radial indications was limited in its ability to scan the entire circumference of the weld. This was due to the limited room between the access cover weld and the shroud and reactor vessel walls. Both areas prevented the fixture from scanning the most susceptible areas to radial cracking. In addition the inspectors noted from the TV monitor which had a camera on the UT fixture, that there are almost no parallel surfaces in the areas adjacent to the wall of the vessel or the shroud. This ledge surface condition would redirect the sound making the detection of radial cracks extremely difficult from the inside of the reactor vessel.

The inspectors noted however that the visual radial indication identified as No.5 was entirely on the 0° shroud access manway cover. The cover has parallel surfaces and there are no scan limitations. Therefore, this indication should be readily detectable with UT. However, the indication could not be detected with the transducer fixture configuration used by GE while the inspectors were at the site.

Review of portions of the 180° access hole cover UT data revealed similar information to the 0° data in that, the UT system easily detected circumferential indications indicative of IGSCC, but did not detect any apparent radial indications.

At the conclusion of the inspectors visit, GE was still adjusting the angles of the transducers in the fixture in an attempt to detect the visual radial indications. The examinations from outside the reactor vessel were not scheduled until the following week.

In addition to observing the ultrasonic examinations on the 180° manway cover and reviewing portions of the data for both the 0° and the 180° manway covers, the inspectors reviewed examiner, equipment, and material qualification and certification records, and reviewed the following procedures to determined if their technical content was adequate and whether they had been properly approved:

Procedure ID & Rev.		Title of Procedure
GE-ADM-1002,	0	Procedure for Review Process and Analysis of Recorded Indications
GE-ADM-1001,	0	Procedure for Performing Linearity Checks on Ultrasonic Instruments
GE-RDE-14-0488	0	Procedure for Inspection and Installation of Access Hole Cover Scanner

GE-UT-211	1	Procedure for Automatic Ultrasonic Examination of the Shroud Support Access Cover Plate
GE-VT-202	0	Invessel Visual Inspection

Within the areas examined, no violations or deviations were identified.

5. Review of TVAs Actions Regarding IN 92-35, " Higher Than Predicted Erosion / Corrosion in Unisolable Reactor Coolant Pressure Bounty Piping Inside Containment at a BWR" (92701)

The inspectors reviewed licensee activities performed to date, and those planned for the near future, with regards to IN 92-35. This review revealed that TVA intended to use the Electric Power Research Institute (EPRI) Checmate Computer Program to select priority ranking for components that would be examined on the feedwater system inside containment in response to IN 92-35.

To date, Pass 1 of Checmate has been performed on the Unit 2 piping to rank each component for examination; however, TVA's site copy of IN 92-35 was missing Attachment 1 to the IN, which is a detail drawing of the piping configuration and the area where the erosion/corrosion was occurring at the site which identified the problem. A review of checmate, pass 1, ranking of the Unit 2 pipe component in question revealed that this component had not been ranked high enough to ensure its examination under TVA's present planning. In addition, a review of TVA's piping drawings revealed that, the Browns Ferry reactors also had the same piping configuration as the plant with the reported condition.

The inspectors discussed the inspection findings with cognizant TVA management and engineering personnel and were assured that the area of concern for IN 92-35 will be examined next refueling outage for Unit 2, and before startup for Units 1 and 3.

Within the areas examined, no violation or deviation was identified.

6. Followup on Generic Letter 88-01, Supplement 1 (92701)

Supplement 1 provided licensee with acceptable alternative staff positions to some of those delineated in GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," dated January 25, 1988. The alternatives are in regard to the inspection of reactor water cleanup system piping outboard of the containment isolation valves; the leak detection requirements pertaining to the operability of leakage measurement instruments; and the frequency of monitoring leakage rates. The supplement also provides clarification or guidance on the staff's positions regarding the sample expansion for Category D welds; the effect of shrinkages resulting from weld overlay repairs or stress improvement on the piping system and its supports and pipe whip restraints; and the technical specification amendments for incorporating the inservice inspection statement and leak detection requirements as delineated in GL 88-01.

TVA is presently working on their submittal GL 88-01, Supplement 1. However, the inspector discussed TVA's proposed positions with the Project Manager for IGSCC mitigation activities at Browns Ferry and obtained a detail status of the IGSCC mitigation activities completed and ongoing for each of the Browns Ferry Units. Basically, TVA's response will be that they intend to meet the original requirements of GL 88-01 at this time. However, TVA may re-evaluate and relax some of their positions when the new standard technical specifications are implemented.

Within the areas examined, no violations or deviations were identified.

7. Review of Pipe Support Calculations for Unit 2

The review of Unit 2 calculations during this inspection is due to the design problems on Unit 3 spring supports reported in Inspection Report Number 50-259,260,296/92-32 as an Inspector Followup Item (IFI) "Design Problems in Spring Supports". The inspector randomly selected 10 pipe support calculations for review from two systems, 01 - Main Steam system and 70 - Reactor Building Cooling Water system. The licensee's General Design Criteria, No. BFN-50-C-7107, Design of Class 1 Seismic Pipe and Tubing Supports, was used for these support calculations since all of them were within the scope of IEB 79-14 program. All five of the supports selected from the Main Steam system were spring can supports. The remaining five supports, from the Reactor Building Cooling Water system, were rigid supports.

The 10 support calculation were partially reviewed and evaluated for thoroughness, clarity, consistency, and accuracy. The review included checks to see that the applied loads used were taken from the latest stress calculation, as well as spring design, member size, weld sizes and symbols, and standard component capacities and settings.

In general, the design calculations were acceptable, except as noted in the "Comments," below. The following table shows the support calculations which were partially reviewed by the inspector.

Support No.	Calculation No.	Rev. No.	System No.
2-47B400S0024	CD-Q2001-881352	1	01

Comment: The spring variability was calculated to be 44%. An inadequate disposition was found to justify the spring variability of 44% instead of the allowable 25%. There was no record of notification of the lead civil engineer found for the calculation.

Support No.	Calculation No.	Rev. No.	System No.
2-47B400S0016	CD-Q2001-881384	1	01

Comments: The spring variability was 31.5%. There was no record of identification to the lead civil engineer found in the calculation.

2-47B400S0012	CD-Q2001-882154	2	01

Comments: The spring variability was 36.4%. There was no record of identification to the lead civil engineer found in the calculation.

2-47B400S0020	CD-Q2001-881417	1	01

Comments: The spring variability was 32%. There was no record of identification to the lead civil engineer found in the calculation.

2-47B400S0027	CD-Q2001-882373	0	01
2-47B464H0035	CD-Q2070-881902	3	70
2-47B464H0029	CD-Q2070-881980	0	70

Comments: The latest stress loads were not incorporated in the support qualification. The angle steel was also not checked for laterally unbraced length per the requirements of Section 1.4.2.12 of General Design Criteria BFN-50-C-7107.

2-47B464S0119	CD-Q2070-881996	2	70
2-47B464R0236	CD-Q2070-883148	1	70

Comments: The latest stress loads were lower than the design loads. The loads were not evaluated and/or the results of evaluation for the impact of the new load change documented in the calculation; which should have happened even though the new loads were lower.

2-47B464S0210	CD-Q2070-883133	2	70

Comments: The angle steel was not checked for laterally unbraced length per the requirements of Section 1.4.2.12 of General Design Criteria BFN-50-C-7107.

Supports No. 2-47B400S0024,...S0016, ...S0012, and ...S0020 had spring variabilities over the 25 percent allowed by Section 1.4.4.1 of General Design Criteria BFN-50-C-7107, Rev. 5. For spring variabilities over 25 percent Section 1.2.2 of the General Design Criteria required that any conflicts or variances shall be identified to the Lead Civil Engineer (TVA) before further action is taken by the designers. The four support calculations listed above did not contain any records of notification to the Lead Civil Engineer. The calculations showed that dispositions of the variances were from the Bechtel pipe support design group to the Bechtel pipe stress analysis group. Section 1.4.4.1 of the General Design Criteria was revised in Rev. 6 to remove the TVA Lead Civil Engineer responsibility for the variances and only require an approval from the stress analyst.

The inspector determined that the disposition of the 44% spring variability, for Support No. 2-47B400S0024, provided by the Bechtel stress analyst, was inadequate. The disposition did not consider an evaluation of the effect of the variability on the pipe stresses or adjacent support load increases.

The failure to follow the procedural requirement to inform the Lead Civil Engineer about the excessive spring variabilities, and the failure to consider the effect of the excessive variability on the pipe stresses and adjacent support loads is considered to be a violation against 10CFR50, Appendix B, Criterion V. Since the licensee has recently revised the General Design Criteria BFN-50-C-7107 to eliminate the requirement to notify the Lead Civil Engineer, and took immediate corrective action to re-disposition the 44% spring variability of Support No. 2-47B400S0024, the problems were considered to have minor safety significance and were not cited because the criteria specified in Section VII.B(1) of the NRC enforcement policy were satisfied. This item will be identified as non-cited violation (NCV) 50-260/92-38-02, "Failure to Properly Identify Support Spring Can Variability".

The inspector found that Support No. 2-47B464H0029 was not based on the loads from latest stress calculation No. CD-Q2070-880983, Rev.2, dated October 27, 1989. The loads used in the support calculation were from Rev.1 of the stress calculation, while the latest loads (from Rev.2) were increased about 43 percent from the original design loads. In addition, the angle steel in this support was not checked for laterally unbraced length per the requirement of Section 1.4.2.12 of the General Design Criteria BFN-50-C-7107. The inspector also found that Support No. 2-47B464R0236 did not contain documentation that is was evaluated for the impact of new load changes, even the new loads were lower. The angle steel on Support No. 2-47B464S210 was also not checked for the laterally unbraced length per the requirement of Section 1.4.2.12 of General Design Criteria BFN-50-C-7107. (The licensee's engineers stated that because the angle size was 4" X 4" X 1/4" and the laterally unbraced length was only 18", the design engineers judged that the allowable bending stress would be 0.6 X yield stress as a normal design condition without stress reduction. It might be true that the unbraced length was relatively short, but it was an improper design practice to estimate the allowable bending stress without checking the unbraced length.)

10 CFR Part 50, Appendix B, Criterion III, Design Control requires that design changes shall be subject to design control measures commensurate with those applied to the original design. TVA Nuclear Engineering Procedure NEP-3.1, Attachment 4, Page 1 of 1 states that design inputs, including information such as loads, temperature ... shall be ... current, referenced, and applied. TVA Rigorous Analysis Checklist requires that the correct support loads from the post processor output, or adjusted loads from hand calculations, have been transmitted to the support designer. The calculation for Support No. 2-47B464H0029 was not revised to reflect the latest stress loads which were 43 percent higher than the design loads. This Item is identified as Violation 50-

260,296/92-38-01, "Inadequate Design Controls for Pipe Support Calculations".

TVA stated that 18 of 38 pipe support calculations contained in stress calculation No. N1-270-2R were reviewed and they had found that this appeared to be an isolated case.

Within the area examined, no deviations were identified.

8. Actions on Previous Inspection Findings (92701)

(Open) Inspector Followup Item (IFI) 50-259,260,296/92-32-01, "Design Problems in Spring Supports"

This IFI involved four concerns related to spring can design. The concerns were: two cold-loads in system 068, inadequate spring cold-load setting in a Torus piping system, two design criteria for Torus and other piping systems, and the Torus piping system not included in the IEB 79-14 program. The inspectors discussed the matters with the licensee's engineers and reviewed the information provided. The licensee agreed to revise the pipe support calculations in system 068 to get one cold-load from the normal operation condition contained in the stress calculation performed by General Electric Company (GE) for the spring cold setting. The calculations will be reviewed during the future inspection.

The licensee explained their position on the remaining concerns as follows:

- It is true that two different design criteria exist for the Torus and other piping systems because they were developed at different times.

The licensee does not plan to combine the two different design criteria into one standard design criteria since it may require more modification work if the design of all Torus piping systems are based on one standard design criteria.

The Inspectors will evaluate the licensee position, in detail, during a future inspection.

Within the areas examined, no violations or deviations were identified.

9. Exit Interview

The inspection scope and results were summarized on November 6, 1992, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report.

With regards to Violation No. 50-260,296/92-38-01, "Inadequate Design Controls for Pipe Support Calculations", the TVA Site Vice President stated he did not agree that this item should be a violation since the mistake did not increase the load calculations for the support and, unless he did not fully understand the problem, this was a isolated example. The inspectors inquired that, if the inspection continued after the exit meeting could the licensee determine that this was an isolated example? The licensee's staff stated that the issue could not be resolved that day. The inspectors informed the licensee the reported item was a violation and that further discussion would only attempt to establish the severity level which is not determined by the inspectors. Therefore, the Vice President comments would be discussed with the appropriate NRC management before severity levels are assigned.

(Open) Violation No. 50-260,296/92-38-01, "Inadequate Design Controls for Pipe Support Calculations", paragraph 7

(Open) NCV No. 50-260/92-38-02, "Failure to Properly Identify Support Spring Can Variability", paragraph 7

(Open) URI No.50-296/92-38-03, "Evaluation of Weld Conditions", paragraph 3

10. Acronyms and Initialisms

ASME - American Society For Mechanical Engineers
BWR - Boiling Water Reactor
B&PV - Boiler and Pressure Vessel Code
E/C - Erosion and Corrosion
EPRI - Electric Power Research Institute
FW - Feedwater System
GE - General Electric
GL - Generic Letter
IGSCC - Intergranular Stress Corrosion Cracking
IN - NRC Information Notice
ISI - Inservice Inspection
NDE - Nondestructive Examination
NRR - Nuclear Reactor Regulations
QA - Quality Assurance
RWCU - Reactor Water Cleanup System
TS - Technical Specifications
URI - Unresolved Item
UT - Ultrasonic Testing
VT - Visual Testing