UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-259/95-52, 50-260/95-52, and 50-296/95-52

Licensee: Tennessee Valley Authority
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1101 Market Street
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Docket Nos.: 50-259, 50-260 and 50-296

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

Facility Name: Browns Ferry Nuclear Power Station Units 1, 2, and 3

Inspection Conducted: August 21-25, and September 11-14, 1995

Inspector: J. J. Lenahan — 10/11/95 Date Signed

Approved by: Jerome J. Blake, Chief — 10/11/95 Date Signed
Material and Processes Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This special, announced inspection was conducted in the areas of Generic Safety Issue (GSI) 40, Pipe Breaks in BWR Scram Systems; modifications of heating, ventilation and air conditioning (HVAC) supports; large bore piping and supports; cable tray and conduit support issues; long term torus integrity; platform thermal growth; moderate energy line break; control rod drive (CRD) piping support modifications; Unit 3 startup issues; and licensee action on previous inspection findings.

Results:

In the areas inspected, violations or deviations were not identified.

One violation was identified for failure to implement installation of a cable tray support in accordance with design drawing requirements - paragraph 3.7.1. An unresolved item was identified regarding performance of walkdown

Enclosure 2

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inspections of the scram discharge volume system piping after the first scram from full power, and operating temperature, following a refueling outage - paragraph 2.

A weakness was identified regarding errors made during the drawing rollup process-paragraph 3.7.1 and 3.7.2.

The following issues are resolved for Unit 3 restart: GSI 40; HVAC support modifications; large bore piping and supports; platform thermal growth, moderate energy line break, CRD pipe support frame modifications, and long term torus integrity.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *T. Abney, Unit 3 Recovery Manager
   R. Gilbert, Operations Procedure Group Supervisor
   #*J. Glass, Acting Lead Civil Engineer
   #*D. Housley, Site Licensing Engineer
   *J. Johnson, Site Quality Manager
   *J. Maddox, Maintenance/Modification Manager
   #*L. Madison, Unit 3 Civil Engineering Supervisor
   #D. Matherly, Operations Supervisor
   #P. Salas, Licensing Manager
   *J. Valente, Unit 3 Engineering Manager
   #*H. Williams, Engineering and Materials Manager
   #*S. Wetzel, Acting Compliance Manager
   #C. Woods, Unit 3 Recovery Field Engineering

   Other licensee employees contacted during this inspection included craftsmen, engineers, technicians, and administrative personnel.

   NRC Resident Inspectors

   *L. Wert, Senior Resident Inspector
   #R. Musser, Resident Inspector
   J. Munday, Resident Inspector

   *Attended August 25 exit interview
   #Attended September 14 exit interview

2. Generic Safety Issue (GSI) 40 - Safety Concerns Associated with Pipe Breaks in The BWR Scram System.

   On January 3, 1986, NRC issued Generic Letter 86-01 which accepted the BWR owners' Group response, BWROG-8420 to resolve GSI 40. BWROG-8420 proposed that leak detection on the Scram Discharge Volume (SDV) system be performed once per refueling cycle by a walkdown within 30 minutes of scram reset after the first scram from full power and temperature following each refueling outage. In an NRC letter to TVA, dated November 7, 1990, NRC listed an outstanding issue regarding GSI 40. The outstanding issue concerned the fact that the licensee's emergency response procedures for Units 1 and 3 did not address the visual inspection of the SDV system within 30 minutes of scram reset after the first scram from full temperature and pressure, following each refuelling outage. In a letter to NRC dated October 1, 1990, Subject: BFN Safety Concerns Associated with Pipe Breaks in the BWR Scram System (GSI 40 and Generic Letter 86-01), the licensee stated that they revised their Unit 2 abnormal operating Instruction (AOI) to require the SDV walkdown to be performed. The licensee also revised Unit 3 Abnormal Operating Instruction 3-AOI-100-1, Reactor Scram, to include this requirement.

The inspector reviewed Revision 5 of procedure 3-AOI-100-1, dated July 10, 1995 and verified the requirement for walkdown inspection of SDV header and instrument volume to check for leakage is specified within the procedure. The requirement is contained in procedure Step 4.2.15.11, which states that a walkdown inspection of the SDV header and instrument "should" be preformed to check for leakage within 30 minutes following scram reset, of the first scram from rated temperature and pressure, after a refueling outage. The inspector discussed the need to revise the procedure by changing "should" to "shall" so that it was clear that the need to perform the walkdown of the SDV is a requirement, and not optional.

The inspector reviewed Unit 2 procedure 2-AOI-100-1, Reactor Scram Revision 40, dated May 12, 1995. Procedure 2-AOI-100-1, also states that the walkdown of the SDV "should" be inspected following scram reset after the first scram from full pressure and operating temperature after a refueling outage. The inspector questioned licensee operations personnel regarding whether they have implemented the required SDV inspection after the first scram from full pressure and temperature during each of the three cycles since the 1991 restart of Unit 2.

The first scrams from full pressure and operating temperature following the refueling outages since the 1991 restart of Unit 2 occurred on August 2, 1991, June 6, 1993, and December 2, 1994. The information provided by the licensee, (Unit 2 Reactor Operator Control room logs for these dates,) was insufficient to determine if the licensee had complied with the commitment to perform the required SDV system inspections in accordance with intent of BWROG-08420. Pending further review by NRC, this issue was identified as Unresolved item 260/95-52-01, SDV System Inspection following Reactor Scram. This is not a Unit 3 restart issue.

Within the areas inspected, violations or deviations were not identified.

3.0 Followup on Unit 3 Restart Issues

3.1 Moderate Energy Line Break

The moderate energy line break (MELB) evaluation was performed to determine the effect of internal plant flooding outside containment from breaks in moderate energy lines (piping) on safe shutdown of the plant. Moderate energy lines are defined as systems with pressures less than 275 psi or temperatures less than 200 degrees F. The Unit 3 evaluation was performed in accordance with the Unit 2 precedent.

The licensee implemented the Unit 3 MELB in a two phase program. Phase I consisted of a detailed drawing review to identify all moderate energy lines which could be sources of flooding in Unit 3, or common class 1 structures; identification of flood compartments/areas; identification of potential drainage paths; and identification of safe shutdown equipment which could be affected by flooding. The Phase II evaluation

included detailed calculations of flow rates from various flood sources; areas affected by flooding; maximum depth of water in the area; and drainage paths from the various flooded areas.

The inspector reviewed the Bechtel Report titled: Moderate Energy Line Break Flood Evaluation Report For Browns Ferry Unit 3, dated April, 1993. This report contains assumptions/conditions for MELB analysis, summary of design methodology, references, design input data, design analysis, and summary and conclusions. The report contained a discussion on Browns Ferry's original methods for conformance to AEC requirements for evaluating flood from MELB, and justification for not including some areas in the MELB analysis. The conclusions of the MELB study were that Browns Ferry conforms to the original licensing basis and that the existing flooding studies and protective measures are adequate. Considerations in the MELB analysis included control of flooding by providing drainage paths from areas containing critical equipment potentially suspectable to flooding, use of curbs/barriers to protect vital equipment, and mounting equipment on pads so that it located above potential flood elevation levels.

The inspector concluded that the licensee's MELB analysis adequately addresses this issue. The licensee's design assumptions and design methodology are technically adequate. The MELB program is acceptable for restart of Unit 3.

3.2 Platform Thermal Growth

The platform thermal growth issue involved the effect of thermal loads on structural steel platforms. During review of structural steel design criteria, the NRC office of Nuclear Reactor Regulation questioned the licensee regarding their use of non-linear analysis which predicted plastic deformation of structures due to thermal loads. As a result of these questions, the licensee performed a comprehensive review of their design criteria and concluded that they would revise the criteria to require steel members to remain within the elastic limit for all loading combinations.

The licensee submitted their revised criteria to NRC for review. A Safety Evaluation Report was issued in a letter to TVA dated December 7, 1993, Subject: Browns Ferry Nuclear Plant Supplement Safety Evaluation of Structural Thermal Growth Design Criteria (TAC Nos. M08618, M80619, and M80620), which accepted the licensee's long-term structural steel design criteria. A followup site audit was conducted by NRR on March 14 and 15, 1995, to review implementation of the licensee's long term design criteria. During this audit, the NRR staff examined completed modifications on Unit 2 platforms required by the revised thermal load design criteria. The results of the audit are summarized in an NRC letter to TVA dated April 20, 1994, Subject: Browns Ferry Nuclear Plant-Audit of Structural Steel Design Criteria Implementation.

The licensee considered thermal loads in their analysis of Unit 3 structural steel platforms. Modifications required by thermal loads included used of slotted holes in beams, addition of beam stiffeners, cover plates, etc. These were implemented as part of design change packages previously inspected by NRC.

Issues regarding modifications to Unit 3 structural steel platforms were closed by the inspector during the inspection documented in NRC Inspection Report number 50-259,260,296/95-41. This issue is resolved for Unit 3 restart.

3.3 HVAC Duct Supports

Numerous discrepancies were identified in 1988 in the design and seismic qualification of heating, ventilation and air conditioning (HVAC) ductwork. The licensee developed a program to inspect safety related HVAC duct work. For Unit 2 restart, the ductwork required for Unit 2 operation was evaluated to an interim operability criteria. A safety Evaluation Report was issued by NRC on August 22, 1990 which concluded that the Unit 2 HVAC ductwork and supports were acceptable for Unit 2 restart. The Unit 2 HVAC ductwork and supports were subsequently qualified for long term requirements of the licensees Design Criteria BFN-50-C-7104.

For Unit 3, a review was performed to identify class I HVAC ductwork which was not previously required for Unit 2 operation. The only areas identified as being specific to Unit 3, were the ductwork associated with the Unit 3 residual heat removal (RHR) and core spray (CS) pump motor coolers. This Unit 3 ductwork and supports were evaluated to the long term requirements of Design Criteria BFN-50-C-7104. A design change notice (DCN) number DCN W28617A, was issued to implement modifications to existing Unit 3 HVAC ductwork supports and added one new support.

The inspector reviewed DCN W28617A and performed a walkdown inspection to examine the new support. The inspector verified the new support, number 3-47B923-11, was installed in accordance with the design requirements shown on drawing number 3-47B923-11, Revision 1. Acceptance criteria for installation of HVAC ductwork supports are specified in Modification and Addition Instruction MAI-4.3, HVAC Duct System, Revision 9, dated June 16, 1993.

The existing-support modifications consisted of increases in the size of some fillet welds to ¼ inch. The inspector reviewed quality control inspection records and weld data sheets for HVAC ductwork support numbers 3-47B923-6, 3-47B923-7, 3-47B923-8, and 3-47B923-10 and verified that the welds were modified as required by the design drawings. This issue is resolved for Unit 3 restart.

3.4 Control Rod Drive Hydraulic Piping System

During inspection of cable tray supports in the Unit 2 reactor building the licensee identified an issue regarding attachment of control rod drive (CRD) system piping to the cable tray support structure. The licensee performed an extensive design evaluation of the Unit 2 CRD piping system and implemented modifications to the Unit 2 CRD pipe support frames. The licensee's Unit 2 CRD pipe frame design reanalysis program was reviewed by NRC during inspections documented in NRC Inspection Report numbers 50-260/89-20,89-31, 89-39, 89-44, 89-62, 90-08 and 92-01. This issue was closed for Unit 2 restart in NRC Inspection Report 259,260,296/90-23.

A walkdown inspection of the Unit 3 CRD pipe support frames showed that the Unit 3 frames, were identical to the Unit 2 CRD frames. Since the Unit 2 frames required extensive modifications, due to cost and schedule considerations, the licensee decided to replace the Unit 3 CRD frames with new supports. The modification which involved installation of 32 new CRD pipe support frames was implemented under DCNs W17652, W17653, and W18645. Three of the new CRD pipe support frames were inspected during the inspection documented in NRC Inspection Report number 259,260,296/95-03.

During the current inspection, the inspector inspected an additional five of the new CRD pipe support frames. The new frames were inspected against the design drawings for configuration, member size, weld size, type and length connection details, and other construction requirements stipulated by the licensee's procedures. CRD pipe support frames inspected were as follows: Support numbers 3-47E-468-102,-103,-104,-106, and -107.

No discrepancies were identified during the walkdown inspection. The inspectors concluded that the modification were implemented in accordance with design requirements. This issue is resolved for Unit 3 restart.

3.5 Large Bore Piping and Supports

The licensee initiated programs in 1979 to comply with IE Bulletin 79-02, Pipe Support Base Plate Design Using Concrete Expansion Anchors, and IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems. Implementation of these programs was delayed by other programs. In addition, the licensee did not include portions of piping systems covered by other programs under the IEB 79-02/79-14 program. Numerous deficiencies were identified by NRC in 1985 and 1986 concerning implementation of these programs. In order to resolve these deficiencies, the licensee made various commitments to NRC regarding improvement to design criteria, reinspection of large bore piping and supports, reanalysis of piping and supports, and implementation of any required modifications. Acceptance of the licensee's design criteria

for analysis of piping and pipe supports by NRC is documented in NUREG-1232, Volume 3, Supplement 1, Safety Evaluation Report for Browns Ferry Unit 2 Restart.

For Unit 3 restart, the licensee has completed all 79-02 and 79-14 work with the exception of a few pipe support modifications on system 10, the reactor head vent. These modifications will be completed after fuel load for Unit 3, but prior to Unit 3 restart.

Inspection of the licensee's 79-02 and 79-14 program for Unit 3 included review of pipe stress analysis, review of pipe support design calculations, and inspection of completed pipe support modifications. These inspections are documented in NRC Inspection Report numbers 50-259,260,296/91-34, 91-42, 92-07, 92-32, 92-38, 93-11, 93-26, 93-29, 94-15, 94-29, and 95-03. This issue is resolved for restart of Unit 3.

3.6 Long Term Torus Integrity

In the early 1980's the licensee implemented a series of modifications to the torus intended to resolve deficiencies identified regarding the original design of the Mark 1 containment system. These modifications involved torus attached piping, and structural reinforcement of the torus and torus related structures. In 1985, discrepancies were identified by NRC during inspections of the as-constructed torus attached pipe support modifications. The licensee's corrective actions included reinspection of the torus attached piping and pipe supports, and reinspection of torus structural modifications and torus related structures. The licensee corrective action for Unit 2 restart were accepted by NRC in Section 2.2.4.4 of NUREG-1232, volume 3, Supplement 2, dated January 23, 1991.

In a letter to NRC dated April 29, 1991, Subject: Browns Ferry Nuclear Plant - Program for Resolving Long-Term Torus Integrity Issue Prior to the Restart of Units 1 and 3, the licensee provided NRC their action plan and commitments for resolution of long term torus integrity for Units 1 and 3. The licensee's corrective actions included walkdown inspections to identify any discrepancies in the torus, evaluation of the discrepancies, and performance of modifications to correct any unacceptable discrepancies.

Inspection of the licensee's implementation of the long term torus integrity program included review of design criteria, design calculations and completed modifications for torus attached piping and pipe supports. These inspection are documented in NRC Inspection Report numbers 50-259,260,296/92-32, 94-15, and 95-03. The inspectors concluded that the installed modifications were acceptable. This issue is resolved for restart of Unit 3.

3.7 Cable Tray and Conduit Supports

Questions were raised by NRC and through the employee concerns program regarding seismic qualification of cable tray and conduit supports. The resolution of this issue can be subdivided into two categories: new cable tray and conduit supports, and evaluation of existing cable tray and conduit supports.

3.7.1 New Cable Tray and Conduit Supports

New supports are those installed since 1986. These supports are designed in accordance with the licensee's seismic design criteria and installed under the licensee's quality assurance program requirements.

The inspector performed a walkdown inspection and examined new cable tray and conduit supports. New supports were inspected against the design drawings for configuration, member size, weld size, type and length, connection details and other constructions requirements. Additional acceptance criteria utilized by the inspector during the walkdown inspection were Modification and Addition Instruction, MAI-3.9, Installation of Cable Tray Cable Tray Supports and Cable Tray Covers, Revision 7; and MAI 3.1, Installation of Conduit and conduit supports, Revision 25.

Cable tray supports examined during the walkdown were: Support numbers 319092-15-B1102-38; 318992-1-B1102-145; 318992-2-B830-56; 318992-3-B1102-32; 318992-8-B1102-174; 3-48B1102-3-21, -38, -122, -145,-164; and W17473-253, -303, and-304.

The following deficiency was noted by the inspector during the walkdown: Two flare-bevel welds on the vertical interfaces of Bill-of-Material items 3 and 4, on cable tray support number 319092-15-B1102-38, had not been completed as required by the design details shown on drawing number DCA W17473-300. Paragraph 6.1.1 of Procedure MAI-3.9 requires cable tray supports to be fabricated and installed according to the applicable design output documents (drawings). The failure of the licensee to install cable tray support number in accordance with the drawing requirements was identified as violation item 296/95-52-02, Failure to Construct Cable Tray Support in accordance with Design Requirements. The licensee issued Problem Evaluation Report (PER) number BFPER 951125 to document and disposition this problem.

The inspector also identified three cable tray supports which had the incorrect support number on the identification tag, (Support numbers 3-48B1102-32, -38, and -145) and one support which had a missing identification tag (Support number 318992-8-B830-15). These items were also included on PER number 1 BFBER 951125. However, since the missing or incorrect tag numbers have no safety significance, these items were not included as part of the violation.

During the walkdown, the inspector also examined a Kellems-grip cable support, support number 3-48B3800-4500, which provides additional vertical support for cables in vertical cable trays. The inspector noted an error in the Bill of Materials (BOM) for Items 1 through 5. The licensee issued BF PER 95-1128 to document and disposition this problem. Further review of this problem disclosed that the error in the BOM for items 1-5 was due to a drafting error in the drawing rollup process. This problem is similar to that identified in Unresolved item 296/95-15-02. Since the support was constructed as required by design requirements, a violation was not identified for this problem; however, the drafting error was identified as a weakness.

The following new conduit supports were also examined during the walkdown inspection: Typical conduit supports number 0-48B805-010, 0-48B805-013, and 0-48B805-014, installed on elevation 593; unique conduit supports number 3-48B3800-2188, -4194, and -4259; and temperature switch support number 3-47B900-212. The conduit supports were constructed in accordance with design requirements; however, an additional drafting error was noted on drawing number 3-48B38004190, Revision 0, regarding incomplete/incorrect drawing notes. This problem was also documented on BFPER 951 128. This was identified to the licensee as another example of the weakness discussed above.

3.7.2 Evaluation of Existing Cable Tray and Conduit Supports

Seismic verification of existing cable tray and conduit supports is being accomplished using the Generic Implementing Procedure (GIP) for Seismic Verification of Nuclear plant Equipment. The GIP was issued by the Seismic Qualification Utility Group (SQUG) in response to NRC Unresolved Safety Issue A-46 (USI A-46), Seismic Adequacy of Mechanical and Electrical Equipment in Operating Plants. The licensee committed to complete the A-46 walkdown for cable tray and conduit supports in Unit 3 prior to restart of unit 3. The inspector reviewed walkdown Instruction CEB-012, Seismic Verification and Assessment of Nuclear Plant Equipment, Revision 0, dated August 17, 1994. This procedure specifies the instructions for implementation of the GIP requirements for personnel qualifications, precautions, methodology, acceptance criteria, and documentation requirements.

The licensee has completed the A-46 walkdown inspections for existing cable tray and conduit supports in all Unit 3 category I structures, except for the drywell. During the A-46 walkdowns, the licensee evaluate cable tray fill, spans, and supports, including anchorage and conduit spans, supports and anchorage using the criteria in GIP. Cable trays, conduits, and supports which did not meet the GIP acceptance criteria were designated as outliers.

The inspector reviewed the results of the licensee's A-46 walkdowns summarized in a walkdown summary table. Outliers are documented on Outlier Seismic Verification Sheets. The outliers are addressed either through a plant work request, or by a design evaluation documented in a calculation using the GIP acceptance and the licensee's design criteria.

Items addressed by work requests included missing or damaged hardware covered by existing plant maintenance procedures. Problems (outliers) which involved questionable design and/or construction practices, e.g. conduit over-spans, apparent inadequate anchorages, potential seismic interactions, supports which do not meet current design practices, etc. were evaluated by the licensee in calculation number CD-Q0000-931227, Revision 1, dated June 8, 1995, Qualification of Cable Tray and Conduit System by A-46 program. Modifications (DCNs) were issued for outlier which could not be qualified.

The inspector reviewed the calculation for completeness, accuracy and adherence to design criteria and procedural requirements. No deficiencies were identified. The inspector walked down three, randomly selected, modifications implemented to resolve A-46 outliers, and verified the modifications were implemented in accordance with the DCN requirements. Modifications examined were as follows:

- Modified Conduit support on drawing number 3-48B3800-3935, Revision 1

- Modified Conduit Support SL No. 3 on drawing number 3-48B3800-3936, Revision 0.

- New conduit support as shown on drawing number 3-48B3800-3938, Revision 0.

No deficiencies were identified.

The inspector conducted a walkdown inspection in the following Unit 3 areas to assess the effectiveness of the licensee's A-46 Cable Tray and Conduit program. Reactor Building Elevations 565, 593, and 621; Control Building cable spreading room and elevation 593 hallway; and Diesel Generator Building. No significant deficiencies were identified, however, several (18) minor items were noted. These included missing or loose lockouts on cable tray hanger rods, loose or mixed conduit clamps, a temporary support still installed in the diesel generator building, broken conduits, and other minor items. The inspector also noted housekeeping deficiencies such as debris in cable trays, missing cable tray covers, and tools and debris left in various areas in the plant. Licensee personnel indicated that the housekeeping deficiencies will be addressed as the work in the areas is completed and the areas turned over to operations for restart.

In the areas examined, the inspector concluded that the licensee's A-46 cable tray and conduit walkdown program meets NRC requirements and is adequate for Unit 3 restart. However, this issue will remain open pending completion of this program in the Unit 3 drywell.

In the areas examined, no deviations were identified.

4.0 Licensee Event Report (LER)

(Closed) LER 259/88-37 Inadequate Design Control Procedures Discrepancies in HVAC Ductwork. In October, 1988 a review of open non-conformance reports identified several discrepancies involving the design and seismic qualification of HVAC ductwork. This LER was closed for unit 2 restart in NRC Inspection Report number 50-259,260,296/91-06.

For Unit 3, a review was performed to identify class I HVAC ductwork that was not required for Unit 2 operation. Modifications were completed as discussed in paragraph 3.3, above. This LER is closed for unit 3 restart.

5.0 Licensee Action on Previous Inspection Findings (92701 and 92702)

5.1 (Closed) Violation Item 259,260,296,85-41-01, Inadequate Design Control for Safety-Related Cable Tray Supports.

This violation was identified in 1985 as a result of a review of design calculation for safety-related cable tray supports in various category 1 structures. Problems identified included improper seismic design analysis of various supports, errors and omissions in the calculations, and failure to perform design verifications. This violation was issued to the licensee on September 8, 1986 as part of a $150,000 civil penalty covering examples of failure to comply with NRC requirements identified in six NRC inspections covering the period from August 12, 1985 through January 31, 1986. The licensee did not contest the civil penalty.

The licensee's corrective actions for this violation are stated in their letter to NRC dated October 8, 1986, Subject: Notice of Violation and Proposed Imposition of Civil Penalty Enforcement Action EA-86-56. The licensee's corrective actions included preparation of procedures and design criteria for design of cable tray supports, using an independent consultant to perform an interim seismic qualification of Unit 2 cable tray supports for Unit 2 restart, and long-term qualification of cable tray supports using the Generic Implementing Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment. The GIP was issued by the Seismic Qualification Utility Group in response to NRC Unresolved Safety Issue A-46 (USI A-46) Seismic Adequacy of Mechanical and Electrical Equipment.

The corrective actions for Unit 2 were reviewed during inspections documented in NRC Inspection Report numbers 50-259,260,296/88-38, 89-21, 89-29, 89-30, 89-32, 89-42, and 89-62. The inspections covered review of design criteria, design calculations, and selected cable tray and conduit support modifications.

5.2 (Closed) Inspector Followup Item 259,260,296,85-51-01, Inspection of Existing Cable Tray Support Systems.

This IFI was identified in 1985 during a followup inspection performed relative to violation item 259,260,296/85-41-01. The inspector noted

that the licensee did not have a written procedure to inspect existing cable tray support systems to assure the as-built cable tray support systems comply with applicable design documents. The licensee is in the process of inspecting existing cable tray support systems, and other mechanical and electrical equipment, using the NRC approved Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, referenced above. The licensee issued walkdown Instruction CEB-012, Seismic Verification and Assessment of Nuclear Plant Equipment, to implement the walkdown program using the GIP. The GIP contains specific requirements pertaining to inspection, evaluation and identification of cable tray support systems which do not meet seismic design requirements.

5.3 (Closed) Unresolved Item 296/86-06-02, Reactor Building Control Bay HVAC Inadequate Design.

This item concern the licensee's identification of inadequate design of HVAC support. This issue was reported to NRC under LER 259/88-037. This unresolved item was closed for Unit 2 in NRC Inspection Report number 50-259,260,296/90-08. For Unit 3, a review was performed to identify the class I HVAC ductwork that was not required for Unit 2 operation. The only areas specific to Unit 3 were the duct work for the Unit 3 RHR and core spray pump motor coolers. The licensee's corrective action for this duct work is discussed in paragraph 3.3, above. URI 296/86-06-02 is closed for Unit 3. URI 259/86-06-02 will remain open for Unit 1.

5.4 (Closed) Unresolved item 296/87-26-03, RHR Pump Suction and Nozzle Load Allowable Are Exceeded.

This item concerned Residual Heat Removal (RHR) Nozzle load allowables as identified by the licensee in deficiency number 87-13-6 of Engineering Assurance Audit 87-13. The licensee revised calculation number CD-Q3073-920014 (System N1-373-5R) and generated new calculation number CD-Q3074-910631 (System N1-374-5R) and CD-Q-3074-910400 (System N1-374-7R) to evaluate the RHR pump suction anchor and nozzle loads. The revised and new calculations qualified the applied loads based on revised design criteria BFN-50-C-7103, General Design Criteria for Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing). The applied loads include I.E. Bulletin 79-14 requirements. The as-built walkdown information and data also were used in the analysis.

The inspectors reviewed the following calculations which qualified the nozzle loads:

- Calculation number CD-Q3074-910631, Revision 4, dated February 27, 1995.

- Calculation number CD-Q3073-920014, Revision 11, dated February 2, 1995.

- Calculation number CD-Q3074-9104000, Revision 5, dated April 5, 1995.

Based on this review, the inspectors determined that the calculations complied with the licensee's design criteria and were acceptable. The piping stresses are within code allowable values. This item is closed for Unit 3. URI 259/87-26-03 will remain open for Unit 1.

6.0 Exit Interview

The inspections scope and results were summarized on August 25 and September 14, 1995, 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. Dissenting comments were not received from the licensee.

Unresolved Item 260/95-52-01, SDV System Inspection Following Reactor Scram, paragraph 2.0.

Violation Item 296/95-52-02, Failure to Construct Cable Tray Support in Accordance With Design Requirements, paragraph 3.7.1.