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TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

JA

6N 38A Lookout Place

October 8, 1986

Mr. James M. Taylor, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

BROWNS FERRY NUCLEAR PLANT - NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTIES - ENFORCEMENT ACTION EA-86-56

This is in response to J. Nelson Grace's letter dated September 8, 1986 which transmitted the subject Notice of Violation and Proposed Imposition of Civil Penalty. Dr. Grace's letter and its enclosure refer to a number of NRC inspections conducted during the August 1985 to January 1986 timeframe that identified areas for which the Browns Ferry Nuclear Plant (BFN) did not comply with NRC requirements, and to the related Enforcement Conference held on May 28, 1986. TVA does not contest the imposition of the civil penalty. My response to the specifics of the BFN enforcement action issues is provided in the enclosure, and my views concerning the need and value of this and other related enforcement matters are addressed by separate correspondence.

Fees in response to the civil penalty of $150,000 are being wired to the NRC, Attention: Office of Inspection and Enforcement.

To the best of my knowledge, I declare the statements contained herein are complete and true.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

S. A. White

S. A. White
Manager of Nuclear Power

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. G. G. Zech
Director, TVA Projects
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

IE: 14

RESPONSE
ENFORCEMENT ACTION REPORT
BROWNS FERRY NUCLEAR PLANT (BFN)
50-259/86-56, 50-260/86-56, AND 50-296/86-56
DR. J. NELSON GRACE'S LETTER TO S. A. WHITE
DATED SEPTEMBER 8, 1986

Violations Assessed Civil Penalties

Item I.A.

Technical Specification 5.6, Seismic Criteria, specifies that station Class I structures and systems are designed to withstand a design basis earthquake. 10 CFR Part 50, Appendix B, Criterion III, Design Control, requires that measures be established to assure that applicable requirements are correctly translated into specifications, drawings, procedures, and instructions and that these measures include provisions to assure that appropriate quality standards are specified and included in design documents. These design control measures must also provide for verifying or checking the adequacy of the design.

Contrary to the above, as of the NRC inspection conducted August 12 - 16, 1985, design discrepancies existed that indicated that some of the cable tray supports in areas of the Control Bay, Diesel Generator and Reactor Buildings, station Class I structures or systems, were not adequately designed to withstand a design basis earthquake and may not have been able to perform their intended function during a seismic event. In addition, a number of design calculations used to qualify many of the typical cable tray supports were not checked or verified.

This is a Severity Level III violation (Supplement I).
(Civil Penalty - $50,000)

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reasons for the Violation

   Inadequate design controls resulted in a failure to coordinate design requirements within and between engineering disciplines. This lack of design control also resulted in engineering calculations which were not adequately prepared nor properly documented.

3. Corrective Steps Which Have Been Taken and Results Achieved

Nuclear Engineering Procedures (NEP) were issued in July 1986. The stringent NEPs place rigorous controls on the design process of cable tray systems in the following areas:

NEP 3.1, Engineering Calculations
NEP 3.2, Design Input
NEP 3.3, Interface Control
NEP 5.1, Design Output
NEP 5.2, Review and Design Verification
NEP 6.1, Change Control

The result of the NEPs is a closely controlled design process which will ensure the adequate design and verification of seismic qualification of cable tray support systems.

In order to ensure that engineering personnel are qualified to perform design activities on cable tray systems, periodic training in all pertinent documents and procedures is required. Individual training records are maintained for each person to ensure that the latest requirements or revisions to documents are made known. Periodic internal audits are conducted to verify training and design compliance.

The following discussion outlines seismic cable tray programs which are being implemented to (1) allow restart and safe operation of unit 2 and (2) provide seismic qualification of cable trays for all three Browns Ferry Nuclear Plant (BFN) units. These two programs have taken into consideration the overfilled cable trays specified in violation I.B.(1.).

1. To verify that cable tray supports are adequate to allow restart and interim operation of unit 2, TVA contracted with United Engineers and Contractors (UE&C) to perform an interim seismic qualification of the unit 2 cable tray system. UE&C issued a formal report containing support qualification calculations and modifications required before unit 2 restart. This report is presently under review by NRC. Those modifications identified will be completed before restart of unit 2.

2. For the long-term resolution of cable tray integrity BFN is planning to use the methodology of NUREG-1030 which was developed to resolve Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Nuclear Power Plants. This method of resolution, jointly developed by NRC and the Seismic Qualification Utility Group (SQUG), uses the results of damage surveys conducted in conventional power plants and industrial facilities which have experienced actual earthquake ground motions.

The surveys show that nonnuclear grade equipment similar to that in nuclear plants, including cable trays, is seismically rugged in general and does not fail under seismic loading. Direct comparison, using qualification criteria jointly developed by NRC and SQUG, will be used to show long-term qualification of BFN cable trays.

TVA contracted with Earthquake Engineering (EQE) to evaluate all BFN cable trays for the long-term resolution of concerns regarding the integrity of cable tray supports under seismic loading. Based on preliminary results of EQE's review, all BFN cable trays would withstand design basis seismic events without modification and continue to function normally.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plan (NPP), Volume 1 provides the actions being implemented at the corporate level to avoid the root cause of inadequate design control. The actions related to the improvement in the design control process for BFN are contained in the NPP, Volume 3. Specific corrective steps for this issue are:

a. The cable tray modifications for unit 2 will be completed using the interim seismic qualification.

b. The long-term qualification of cable trays will be completed using the methodology of NUREG-1030 to resolve Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Nuclear Power Plants.

5. Date When Full Compliance Will Be Achieved

Seismic qualification (including any modifications required) of the unit cable tray systems or common cable tray systems needed to support the units will be accomplished before their respective startup dates.

Item I.B. (1 through 5)

10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that measures be established to assure that conditions adverse to quality are promptly identified and corrected. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to the appropriate levels of management.

Contrary to the above, the licensee failed to take adequate measures to assure conditions adverse to quality were promptly identified and corrected in the following circumstances:

Item I.B.(1)

On February 18, 1981, the licensee identified a significant condition adverse to quality and initiated Corrective Action Report (CAR) 81-035 that addressed overfilled cable trays and cable penetrations in the cable spreading rooms. Various actions were taken from February 1981 until July 1985, none of which succeeded in correcting the overloaded condition of the cable trays.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reasons for the Violation

   The contributing factor was that the importance of the CAR program was not always recognized by line management. As a result, inadequate responses were frequently received to disposition the problem documented. The inadequate responses required additional attention by QA personnel. This additional attention was often delayed because of inadequate staffing. For most of the period that CAR 81-035 was being handled, BFN Quality Assurance (QA) Staff had one person assigned part-time to the CAR program. Requests for additional personnel did not receive the necessary attention of higher management.

The combination of inadequate staffing and inadequate responses to CARs resulted in some CARs remaining open for long periods of time, as was the case of CAR 81-035.

3. Corrective Steps Which Have Been Taken and Results Achieved

   a. Corrective Action Reports

      Management is committed to resolving CARs expeditiously. CARs are discussed in daily Quality Assurance Staff meetings, and the Site Director meets periodically with the Site Quality Manager and

quarterly with the Corporate Director of Nuclear Quality Assurance to discuss CAR status and escalate corrective actions. Managers at all levels are being held accountable for timely resolution of CARs in their area of responsibility. The site QA organization has been expanded to better track and evaluate CARs. A Quality Assurance Technical Services Staff has been established and is responsible for tracking the issuance, verification, and closure of CARs. The staff is assigned the responsibility of preparing monthly reports to management to keep them apprised of the status of plant-initiated CARs.

b. Cable Tray Filling

Corrective actions have been taken to preclude future overfilling of cable trays. Stringent NEPs were issued in July 1986 which place rigorous controls on the design process of new cable loaded in existing or new cable trays in the following areas:

NEP 3.1, Engineering Calculations
NEP 3.2, Design Input
NEP 3.3, Interface Control
NEP 5.1, Design Output
NEP 5.2, Review and Design Verification
NEP 6.1, Change Control

The result of the NEPs is a closely controlled design and training process which will ensure the adequate design and verification of cable loading in existing or new cable trays.

It was decided to resolve the overloading by qualifying the cable tray supports. CAR 81-035 stated that an effort was required to qualify cable tray supports that are already in existence and loaded with cables. Two programs, discussed in detail in our response to Violation 1A, outline seismic programs which are being implemented to (1) allow restart and safe operation of unit 2 and (2) provide long-term seismic qualification of cable trays for all three BFN units.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plant (NPP), Volume 1 provides the actions being implemented at the corporate level to avoid the root cause of inadequate design control. The actions related to the improvements in the design control process for BFN are contained in the NPP, Volume 3. Specific corrective steps for this issue are:

a. BFN is committed to completing the cable tray modifications for unit 2 using the interim seismic qualification.

b. BFN is committed to completing the long-term qualification of cable trays using the methodology of NUREG-1030 to resolve Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Nuclear Power Plants.

5. Date When Full Compliance Will Be Achieved

Seismic qualification (including any modifications required) of the unit cable tray systems or common cable tray systems needed to support the units will be accomplished before their respective startup dates.

Item I.B.(2)

The licensee discovered on August 14, 1985, that the corrective action taken in response to a previous Notice of Violation (NOV) (Inspection Report 50-259, 50-260, 50-296/84-23) had not been accomplished. Mechanical Maintenance Instruction (MMI)-6 implements the Browns Ferry Technical Specification surveillance requirement 4.9.A.1.d for routine diesel generator inspections. The licensee committed in response to the NOV to revise MMI-6 to include, by October 5, 1984, the manufacturer's inspections/maintenance recommendations given in Electro-Motive Division Maintenance Instruction MI-1742, Revision E; Scheduled Maintenance Program 999 System Generating Plants. The recommended six and twelve year maintenance intervals were neither included in the revised instruction nor was an evaluation performed to determine whether or not the intervals should have been included in the instruction.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reason for the Violation

   The failure was apparently due to an oversight on the part of the Mechanical Maintenance Supervisor and the Mechanical Maintenance Section cognizant engineer.

3. Corrective Steps Which Have Been Taken and Results Achieved

   Following revision of the Mechanical Maintenance Instruction detailing the scheduled maintenance program for the diesel generators, implementation of the replacement components began in October 1985. All BFN diesel generator recommended maintenance had been accomplished as of April 6, 1986. The manufacturer's multiyear inspections have been incorporated into the plant's preventive maintenance program.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

   The Nuclear Performance Plan (NPP), Volume 1 provides the actions being implemented at the corporate level to avoid the root cause of inadequate maintenance practices. The actions related to the improvements in the maintenance process for BFN are contained in the NPP, Volume 3. Specific corrective steps for this is that BFN management has and will continue to stress the importance of proper attention to component replacement or maintenance, including performing the diesel generator recommended maintenance. Placement of the multiyear inspections in our formal preventive maintenance program should prevent future violations.

5. Date When Full Compliance Will Be Achieved

   TVA is in full compliance. The manufacturer's multiyear inspections have been incorporated into the plant's preventive maintenance program.

Item I.B.(3)

A Notice of Violation (Inspection Report 50-259, 50-260, and 50-296/85-28) involved incorrect service mounting of the diesel generator battery racks. The licensee responded to the Notice of Violation and indicated that the battery racks were seismically qualified. During battery rack maintenance on April 20, 1985, four foundation mounting studs were broken that were replaced by April 22, 1985. A metallurgical evaluation was requested by the licensee to evaluate the failure mechanism of the four studs. The evaluation revealed that the stud material was unacceptable for welding; thus, the battery racks were seismically unqualified. Corrective action was not taken and the condition was not reported to appropriate levels of management until September 24, 1985, when the diesel generators were declared inoperable.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation. Additional details are provided in item 2.

2. Reason for the Violation

   Inspection Report 85-28 involved a seismicity problem with the mounting details of the diesel generator battery racks. The corrective action was to install shims to restore seismic qualification. During this repair process, an unrelated condition involving the stud configuration on the battery rack mounting rack was encountered which ultimately caused the seismic qualification to again be revoked in September 1985. Details of the problem follow below.

   The repair process to shim the diesel generator battery racks involved physical removal and reinstallation of the racks. During the reinstallation work, four anchor stud welds separated from the embedded base plate while torquing the holddown nuts. There are 64 studs altogether for eight battery racks. This problem was attributed to weld deficiencies during the original installation of the battery racks. Weld repairs were made to the four studs and the battery racks put back in service in April.

   In mid-June 1985, the repair package was transmitted to quality assurance (QA) for a detailed postwork review before lifetime storage. In early July 1985 the QA review noted minor discrepancies with the repair package and issued a Discrepancy Report to document their findings. One of their findings was that the repair procedure did not specifically reference the applicable stud material specification for the stud to baseplate weld procedure.

The acceptability of the welds was not in question in this timeframe since the weld procedure had been reviewed. To fully close out the work package and Discrepancy Report, it was thought prudent to verify material traceability in the original installation package of the battery racks (circa 1975). The installation package was located and reviewed in early August but did not contain the material certification for the studs. The alternative was to actually take samples of the stud material for metallurgical analysis. The samples were taken in August and sent to offsite laboratories for analysis. The final results of the analysis were transmitted to the site in late September 1985. The analysis showed the stud material was not the material specified in the design drawings and that the material was not suitable for welding application. This conclusion affected all 64 studs. The battery racks could not be demonstrated to be seismically qualified and the diesels were declared inoperable on September 24, 1985. As can be seen by the preceding explanation, the timeliness of corrective action was lengthened by the delayed recognition of the fact that a material deficiency existed, rather than welding deficiencies which had been repaired.

3. Corrective Steps Which Have Been Taken and Results Achieved

The diesel generators were immediately declared inoperable (maintained available) due to the nonconforming material. Corrective action to replace the stud bolts with a suitable type was promptly initiated. The diesel generator battery rack stud bolts were replaced in a planned sequence and completed by October 24, 1985.

Current procedures for modifications and additions to the facility are rigorous in terms of work control and documentation for requirements, such as material certification and traceability. These procedures minimize the likelihood of recurrence of this type material problem.

4. Corrective Steps Which Will Be Taken To Avoid Further Violations

The Corporate Nuclear Performance Plan (CNPP), Volume 1 provides the actions being implemented at the corporate level to correct the root cause concerning poor control and timeliness of corrective action. The actions related to the improvements in the corrective action control process for BFN are contained in the Browns Ferry NPP, Volume 3. Additional corrective steps for this issue are that Standard Practice BF-7.6, Attachment F, Standard Guidelines for the Preparation/Review of Maintenance Requests (MR), will be revised to include a requirement to list the base material specifications for safety related welding repairs. The responsible section will include the material specification when preparing the MR package that requires welding, and QA will ensure the material is acceptable when they perform their independent review.

5. Date When Full Compliance Will Be Achieved

Full compliance will be achieved by November 30, 1986, which is the anticipated date of the implementation of revised Standard Practice BF-7.6.

Item I.B.(4)

During the inspection conducted October 26 - November 20, 1985, it was found that the licensee had not taken action to correct a significant condition adverse to quality. The FSAR, Section 8.5.5, specifies that maintenance on the diesel generators be conducted in accordance with manufacturer's recommendations. A vendor diesel generator newsletter (Power Pointer) dated November 28, 1979, indicated that any viscous crankshaft vibration damper bearing with a 1969 or earlier serial number should be removed from service immediately and returned to the manufacturer due to potential for failure. However, the licensee never acted upon the newsletter although this type of damper was installed in the plant diesel generators. In addition, notwithstanding the FSAR, the licensee has no program to review vendor recommended equipment modifications to assure diesel reliability.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reasons for the Violation

   In June 1980 purchasing contract 81P69-302736 was initiated to procure the recommended replacement type dampers. The plant maintenance staff planned to replace the dampers installed on the diesel generators upon receipt of material. However, as the replacement dampers were of a fundamentally different design, Design Change Request (DCR) 2547 was initiated in January 1981 to replace the viscous damper with the improved design gear type damper. The DCR package was subsequently reviewed and was assigned a routine priority. Modifications for which no formal commitment existed were given lower priority during the modification review process. Final issue of drawings for DCR 2547/Engineering Change Notice P0570 was May 1, 1984. The proper attention was not given to the diesel generator newsletter.

3. Corrective Steps Which Have Been Taken and Results Achieved

   Implementation of the replacement components began in October 1985. All BFN diesel generator dampers had been replaced as of April 6, 1986.

   Various vendor information letters are routinely reviewed by plant section supervisors, as addressed in Standard Practice BF-21.17, which stipulates that the assigned section supervisors are responsible for their review and subsequent recommended actions.

The recent increase in the design staff located at the site, and a formal improved priority ranking system for DCRs has significantly decreased the time required to request and implement priority modifications of this nature.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

BFN management has and will continue to stress the importance of proper attention to vendor information letters or instructions requiring component replacement or maintenance. In addition, as is emphasized in the Corporate NPP, Volume 1, several measures are in process to enhance operating experience review and vendor information control systems. The actions related to the improvements in the operating experience review area relating to pertinent vendor information at BFN are contained in NPP, Volume 3. These will provide a method to ensure that all pertinent vendor information is reviewed, tracked, and appropriate corrective actions are taken in a timely manner. We will review the progress of these enhancements and evaluate if further or interim measures are necessary to ensure timely and correct responses to safety significant vendor information bulletins.

5. Date When Full Compliance Will Be Achieved

TVA is in full compliance. The diesel generator damper replacements were complete on April 6, 1986.

Item I.B.(5)

An emergency design change request (DCR 2675) was written in 1981 for the replacement of the diesel generator turbocharger drive gears. The turbocharger drive gears were known to be inadequate for no load/light load operations and failure could occur if the engine was operated in excess of 200 hours at less than 20 percent load. Although this condition adverse to quality had been corrected by the Sequoyah Nuclear Plant (Supplement 2, NUREG-0011, August 6, 1980), as of the time of the inspection on October 26 - November 20, 1985, the change had not been implemented by Browns Ferry. The replacement was completed on April 6, 1986.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation in that a condition adverse to quality, the manufacturer's recommended replacement of the standby diesel generator turbochargers, was not accomplished in a timely manner.

2. Reasons for the Violation

   DCR 2675 was initiated in late 1981 to accomplish the turbocharger replacement. Caution statements were added to plant instructions to reduce, whenever possible, no-load/light-load operation. The DCR was assigned a routine priority since total diesel generator operating hours in the light load/no load mode were believed to be well below the minimum noted failure time of 200 hours. However, in May 1982 the DCR was placed on hold as no formal commitment existed to complete the turbocharger changeout. Modifications for which a formal commitment existed were given higher priority during the modification review process.

3. Corrective Steps Which Have Been Taken and Results Achieved

   Following a review of all existing DCRs related to the diesel generator system in November 1984, it was determined to accomplish the component replacement as a maintenance item. All BFN diesel generator turbochargers were replaced with the recommended replacement featuring the heavy duty drive gear train as of April 6, 1986.

   Vendor information letters or instructions are routinely reviewed by plant section supervisors and addressed in Standard Practice BF-21.17 which stipulates that the assigned section supervisors are responsible for their review and subsequent recommended actions.

The recent increase in the design staff located at the site, and a formal improved priority ranking system for DCRs has significantly decreased the time required to request and implement priority modifications of this nature.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

BFN management has and will continue to stress the importance of proper attention to vendor information letters or instructions requiring component replacement or maintenance. In addition, as is emphasized in CNPP, Volume 1, several measures are in process to enhance operating experience review and vendor information control systems. The actions related to the improvements in the operating experience review area relating to pertinent vendor information at BFN are contained in NPP, Volume 3. These will provide a method to ensure that all pertinent vendor information is reviewed, tracked, and appropriate corrective actions are taken in a timely manner. We will review the progress of these enhancements and evaluate if further or interim measures are necessary to ensure timely and correct responses to safety significant vendor information letters.

5. Date When Full Compliance Will Be Achieved

TVA is in full compliance. The replacement featuring the heavy duty drive gear train was completed on April 6, 1986.

Item I.C. (1 through 4)

10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures and Drawings, requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and that these activities be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, the licensee failed to prescribe adequate instructions, procedures, or drawings to implement such instructions, procedures, or drawings in the following circumstances:

Item I.C.1.

During surveillance testing on November 19, 1985, the licensee discovered that the 250 volt DC control power to the 4160 volt shutdown board "A" was not connected per Plant Drawing 0106D8860. The electrical supply cables for the normal control power supply (250 volt DC shutdown board "A" battery) and the alternate control power supply (250 volt DC battery board "2" to the unit 2 battery) were reversed. This wiring error resulted in control power being supplied from a 30-minute rated source in lieu of the required 3-hour rated source. This condition may have existed since 1973.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reason for the Violation

   A wiring error was identified by TVA that reversed the normal and alternate power feeder cables to the 4160 volt shutdown board "A" control power bus. However, the allegation that the wiring error resulted in control power being supplied from a 30-minute rated source instead of a three-hour rated source is incorrect. Both the normal and alternate control power sources are 30-minute rated sources in accordance with design requirements (reference Final Safety Analysis Report (FSAR), section 8.6). Although the design requirements for the normal and alternate control power batteries specify 30-minute rated sources, all the shutdown board batteries are conservatively sized and can supply their required loads for three hours. TVA believes this is an important fact in evaluating the safety significance of the violation.

   The reason for the violation was an error in the installation and testing of the modification in 1973 which added the dedicated control power battery to 4kV shutdown board "A." Personnel involved in the modification in 1973 did not give the proper attention to the installation or testing details.

3. Corrective Steps Which Have Been Taken and Results Achieved

Significant improvements have been made in the modification process since 1973 when the failure to implement the wiring modification occurred. The present modification procedure requires detailed instructions, workplans, and inspection records including quality control holdpoints before commencement of the modification.

The wiring error was discovered on November 11, 1985 during lineup of board control power for performance of the battery surveillance procedure on shutdown battery "A." Selecting the alternate source and beginning isolation of the normal source caused a temporary loss of control power to 4kV shutdown board "A." Control power was restored, and the surveillance was immediately stopped. An investigation was initiated. Wiring to the normal and alternate sources was found reversed at the shutdown board. The wiring error was corrected on November 22, 1985. Physical inspections of the other 4kV shutdown boards with the same type control power arrangements were conducted, and a review of the drawings and documentation was performed for the modification which added the dedicated batteries. No other wiring errors were found on the other boards.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plant (NPP), Volume 1 provides the actions being implemented at the corporate level to alleviate the root cause of inadequate control of modifications. The actions related to the improvements in the modification control process for BFN are contained in the NPP, Volume 3.

The detailed instructions, the workplan reviews, the inspection records, and the post modification testing documentation requirements which exist today preclude recurrence of this problem. The present policies of encouraging employees to question and report apparent discrepancies (as seen during the November 11, 1985 battery surveillance immediate stop work) also help preclude recurrence.

5. Date When Full Compliance Will Be Achieved

Full compliance was achieved on November 22, 1985 when the wiring error was corrected.

Item I.C.(2)

During the inspection conducted November 21 - December 31, 1985:

a. The licensee discovered that the units 1 and 2 diesel generators oil pressure switches PS-82-29 A, B, C, and D (MB-3) were not functional. The oil line connection block was not ported (drilled out) in accordance with Plant Drawing 45N767-4 for the oil line connecting the pressure switches. This prevented the pressure switches from functioning as a backup to the speed sensor to prevent engagement of the number two bank of air start motors if oil pressure was not maintained after the diesel started.

b. NRC discovered that Plant Operating Instruction OI-82 for the diesel generators was not appropriate to the circumstances because it did not address the function of the Low Low Lube Oil Pressure Light. Because there was no oil pressure to the pressure switch when the engines were running, no oil pressure in the sensing line should have caused illumination of the Low Low Lube Oil Pressure Light next to the emergency stop button on the diesel generator control board in the control room. However, all four of the lights in both units were not functional.

1. Admission or Denial of the Alleged Violation

TVA does not contest the violation.

2. Reasons for the Violation

The reason for violation I.C.(2.a.) was improper installation and testing. The installation occurred during the construction phase before 1977. The preoperational test apparently failed to identify the problem.

Violation I.C.(2.b.) was the result of a procedural deficiency in that the function of the low low lube oil pressure light is not addressed. Thorough procedure review in conjunction with proper attention to the indicating light should have identified this deficiency.

The subject switches provide two functions: (a) disengagement of the No. 2 air start circuit and (b) control room indication of low low oil pressure.

a. Disengagement of No. 2 starting circuit - The pressure switch can either fail open or fail closed. A fail open condition would prevent the No. 2 air start circuit from starting the diesel; however, this would not prevent in any way the No. 1 air start circuit from starting the diesel. A fail closed condition (existing configuration) would not prevent the No. 2 air start circuit from

starting the diesel. It would prevent the switch from disengaging the No. 2 air start circuit upon diesel oil pressure reaching 30 psig; however, a speed sensing switch disengages this circuit when the diesel speed reaches 125 RPM. Therefore, a failure of the pressure switch does not prevent the diesel generator from starting.

b. Indication - A fail open condition would cause the loss of low low oil pressure indication in the control room when the diesel is running. A fail closed condition should cause the indicating light to remain lighted continuously while the diesel is running.

OI-82 did not address this light; therefore, no operator action would be taken in the event low oil pressure was thus indicated. However, the control room has a separate indication of low oil pressure (annunciator). Different switches provide inputs to this annunciator.

An unresolved safety question determination approved on November 29, 1985 documented that an unreviewed safety question (USQ) was not involved in this problem. OI-82 does address the low oil pressure annunciator located in the main control room. This annunciator alarms if the diesel speed has been greater than 125 RPM for two minutes and if either of the following two conditions is met.

(1) Oil pressure at the engine instrument manifold is less than 20 psi if the diesel speed is between 125 RPM and 870 RPM.

(2) Oil pressure at the engine instrument manifold is less than 44 psi if the diesel speed is above 870 RPM.

In the event the annunciator alarms, OI-82 contains instructions for the operator to follow; therefore, the operator has sufficient information available in the event a low oil pressure event occurs.

In summary, this self-identified deficiency would not prevent the diesel generators from starting or prevent the low oil pressure from being identified due to the annunciator. TVA believes these are important facts in evaluating the safety significance of the violation.

3. Corrective Steps Which Have Been Taken and Results Achieved

This condition had not been detected by BFN personnel in the past because the pressure switch was periodically tested by disconnecting the oil line at the pressure switch. The condition was found by BFN personnel by pressurizing the oil line upstream of connection block with no observed response from the pressure switch. This condition was found to be common to all four unit 1 and 2 diesels. The unit 3 diesels were not affected as the oil lines were connected by a series of "T" connections. The condition has been corrected, and a complete test, which included the control room indicating lights, has been performed for the unit 1 and 2 diesel generators. The subject switches are now properly functional.

Annunciator Response Procedure, Panel 9-23, is currently in Document Control for processing. It includes identification of the low low lube oil pressure light and contains instructions for the operator to follow in the case of illumination of the light. The operators have been trained in the instructions to follow in the case of illumination of the light.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plan (NPP), Volume 1 provides the actions being implemented at the corporate level to alleviate the root cause of inadequate emphasis to maintenance and surveillance procedures. The actions related to the improvements in the maintenance and surveillance procedural process for BFN are contained in the NPP, Volume 3.

Operators are continually reviewing annunciator response procedures for inclusion of any detail which would further improve operational performance.

5. Date When Full Compliance Will Be Achieved

Full compliance will be achieved November 30, 1986 which is the anticipated date for issuance of Annunciator Response Procedure, Panel 9-23.

Item I.C.3

Plant Instruction SMI 192.2, Local Power Range Monitor (LPRM) Maintenance Instruction, was inappropriate to the circumstances for the LPRM changeout conducted in unit 2 on November 20, 1985. The instruction did not address the abnormal operation for a LPRM changeout with a LPRM tip broken in the LPRM tool or a LPRM stuck behind a source pin rack. Continued operation resulted in the "hot" tip of the LPRM breaking the water surface of the spent fuel pool, creating excessive radiation levels in the area and an unnecessary radiation hazard to personnel.

1. Admission or Denial of the Alleged Violation

   TVA does not contest the violation.

2. Reasons for the Violation

   Operations personnel were transporting an old LPRM from position 16-49 in the unit 2 reactor to the spent fuel pool. While being dragged across the edge of the vessel flange, the tip of the spring (hot) end of the LPRM was pulled off. The tip and LPRM remained in the LPRM tool. This allowed the LPRM to slide through the LPRM hook, preventing a cinch hold. The LPRM safety hook had been installed before this event. After the LPRM was placed in the spent fuel pool, the cold end was secured to the southeast corner of the spent fuel pool. Attempts were made to create a downward bow on the LPRM to allow storage. The assistant unit operator (AUO) attempted to remove the tool, but the sliding effect of the LPRM through the tool hampered efforts to store the LPRM. During this movement of the LPRM tool, the LPRM was caught behind the source pin rack. The AUO attempted to remove the LPRM from behind the source pin rack by overriding the limit switch on the monorail hoist. This permitted him to bring the LPRM closer to the pool surface than five feet. The monorail hoist and LPRM tool were abandoned at this point after returning the tool five feet below the water surface of the spent fuel pool. Manipulations now continued using J-hooks and the LPRM safety hook to get the LPRM placed properly. During this phase of the operation, trying to free the LPRM from behind the rack of source pins, the AUO and boilermakers pulled the LPRM closer than five feet to the surface of the water. At this time, the health physics (HP) technician was located on the south side of the spent fuel pool monitoring the surface of the water over the middle (highest dose rate) of the LPRM. The boilermaker foreman and HP technician noted the LPRM hot end had risen to about 18 inches below the water's surface. The HP technician relocated to the west end of the spent fuel pool to monitor the LPRM hot end. The HP technician noted a dose rate increasing to two rems/hour.

The following actions occurred almost simultaneously:

a. The HP technician turned to inform Operations to stop raising and to lower the LPRM.

b. The Ludlum 300 ARM on the refueling bridge and the ARM on the north wall (see Attachment 2) of the refueling floor alarmed.

c. The LPRM was lowered by Operations.

The above events occurred sufficiently fast that the operator lowered the LPRM before the HP technician could verbally transmit the direction to stop raising and lower the LPRM. The HP technician estimated the event to last about two seconds. The increased dose rate initiated secondary containment isolation and standby gas treatment. The alarm returned to normal (cleared immediately upon lowering the LPRM).

Root Cause of the Incident:

a. The use of a procedure which did not minimize the potential for physical damage to the LPRM which resulted in increased difficulty in moving the LPRM.

b. A prejob briefing which did not include all aspects of the operation.

c. An unanticipated and unplanned configuration with the damaged LPRM caught behind the source pin rack.

The greatest exposure received by an individual was 30 millirem. TVA believes this is an important fact in evaluating the safety significance of the violation.

3. Corrective Steps Which Have Been Taken and Results Achieved

The LPRM procedure was reviewed by all responsible sections and revised to incorporate the following changes:

a. Additional radiological caution statements added.

b. A statement added to require a formal operational briefing before the start of work.

c. The method of physically moving the LPRM was revised to preclude the recurrence for future similar events.

d. Personnel responsible for this task have been informed of the corrective actions and have received a critique on the incident.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plant (NPP), Volume 1 provides the actions being implemented at the corporate level to alleviate the root cause of inadequate emphasis on the conduct of operations. The actions related to the improvements in the conduct of operations for BFN are contained in the NPP, Volume 3.

A formal operational briefing will be required before the start of work. The briefing will review in detail the steps necessary to accomplish each LPRM changeout without necessary difficulties or hazards. Corrective actions as a result of the LPRM changeout on November 20, 1985, will be emphasized.

5. Date When Full Compliance Will Be Achieved

TVA is in full compliance.

Item 1.C.(4)

The licensee failed to ensure that design drawings referenced the correct design specifications. On October 22, 1985, the licensee discovered four design drawings (730E918, Engineered Safeguards; 730E915, Reactor Protection System; 730E930, Core Spray; and 730E927, Primary Containment Isolation) referenced design specification 22A1421 which is not applicable to Browns Ferry instead of the correct specification 22A2809. This specification defines the criteria for the separation and identification of reactor safeguards electrical equipment.

1. Admission or Denial of the Alleged Violation

TVA does not contest the violation.

2. Reasons for the Violation

TVA concurs that four BFN active design drawings, and one superseded design drawing referenced an incorrect design specification. They are as follows:

730E918, Engineered Safeguards
730E915, Reactor Protection System
730E930, Core Spray System
730E927, Primary Containment Isolation
730E921, Superseded by 45N626

The following chronological listing should explain the reasons for this finding:

On July 19, 1968 General Electric (GE) issued generic separation criteria 22A1421, revision 0, while TVA was still reviewing the draft specification. TVA was never aware that this specification was issued.

On December 20, 1968 TVA returned their comments on the generic separation criteria (draft 22A1421) to GE to be resolved for BFN.

On May 7, 1969 GE issued drawing 730E915-1, revision 0, for BFN with the general specification (22A1421) referenced.

On June 5, 1969 GE issued drawing 730E918-1, revision 0, for BFN with the generic specification (22A1421) referenced.

On August 18, 1969 GE submitted separation criteria 22A2809, revision 0, to TVA. This criteria is a BFN unique separation criteria.

On September 22, 1969 GE issued drawing 730E927-1, revision 0, for BFN with the general specification (22A1421) referenced.

On October 8, 1969 GE issued drawing 730E930-1, revision 0, for BFN with the generic specification (22A1421) referenced.

On March 25, 1970 TVA approved "with corrections noted" the BFN unique separation criteria 22A2809, revision 0.

On October 19, 1970 GE submitted separation criteria 22A2809, revision 1, for review.

On December 1, 1970 TVA approved 22A2809, revision 1.

On June 9, 1972 GE submitted separation criteria 22A2809, revision 2, for review.

On September 25, 1972 TVA approved 22A2809, revision 2.

From this listing it can be seen that GE issued separation criteria 22A1421 while TVA was still reviewing the draft specification, and that the drawings in question were issued for BFN by GE referencing the generic separation specification (22A1421) during the same time period. In reviewing the drawings, TVA did not find the discrepancy because TVA reviewed just for interfaces, not for total content and design verification. Although TVA approved the BFN unique separation criteria and its revisions, GE never revised their drawings to incorporate 22A2809 for BFN.

3. Corrective Steps Which Have Been Taken and Results Achieved

Significant Condition Report (SCR) BFNEEB8606, revision 1, was written to document and track this condition adverse to quality. An engineering report/failure evaluation was performed on this SCR which documents the reviews of the design specifications (one review performed by TVA and the other review performed by GE). These identify the differences between the design specification and resolve the differences. The conclusion to this report states that "each system affected by the differences in the incorrect referenced design specification 22A1421, revision 1, and the correct design specification 22A2809, revision 2, has been designed and installed consistent with the intent of design specification 22A2809, revision 2." With regard to the safety significance of this issue, the integrity of separation requirements as applied to plant design has not been compromised.

4. Corrective Steps Which Will Be Taken to Avoid Further Violations

The Nuclear Performance Plan (NPP), Volume 1 provides the actions being implemented at the corporate level to avoid the root cause of inadequate design control. The actions related to the improvements in the design control process for BFN are contained in the NPP, Volume 3.

An engineering change notice was issued to correct the wrong design specification on the drawings identified, thus removing the potential of the wrong design specification being used on future design modifications. Corrections of the design specification references were completed on June 1, 1986 and the active drawings reissued.

To evaluate for generic implications, TVA implemented a sampling program on the residual heat removal system to review the conceptual drawings for design specifications and ensure that the design specifications identified from the drawings are the correct design specifications for BFN. The sampling program was completed on July 1, 1986, and no discrepancies were noted; thus, the incorrect references were deemed to be an isolated event.

5. Date When Full Compliance Will Be Achieved

TVA is in full compliance.

Violation Not Assessed a Civil Penalty

Item II

10 CFR 50.59(a) allows the holder of a license to make changes in the facility as described in the Final Safety Analysis Report (FSAR) without prior Commission approval unless it involves a change to the Technical Specifications or is an unreviewed safety question. An unreviewed safety question is created if the consequences of an accident or the malfunction of the equipment important to safety previously evaluated in the FSAR may be increased.

10 CFR 50.59(b) requires that the licensee maintain records of changes in the facility to the extent that such changes constitute changes in the facility as described in the FSAR. These records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

Contrary to the above, the licensee changed the facility as described in the FSAR without having determined whether the change involved an unreviewed safety question. On November 25, 1985, the licensee changed the acceptable closure time of secondary containment isolation dampers from two seconds as specified in Section 5.3.4.2 of the FSAR to 10 seconds. Although the licensee performed an analysis, it did not determine whether the change involved an unreviewed safety question. The change was implemented by the licensee through an internal memorandum pending a change to the FSAR to be submitted as part of the routine annual update per 10 CFR 50.71.

This is a Severity Level IV violation (Supplement I).

1. Admission or Denial of the Alleged Violation

   TVA disagrees that a violation of 10 CFR 50.59 occurred. We believe the situation regarding isolation damper timing did not constitute a change in the facility or a change to any procedure within the scope of 10 CFR 50.59. Our justification for this position is detailed below.

   Secondary containment at BFN is divided into three reactor zones and a common refueling zone. Each individual zone has sets of paired redundant supply and exhaust isolation dampers. Radiation monitors are installed in the vicinity of the exhaust dampers which will automatically isolate the dampers and start the standby gas treatment system (SGTS) on high radiation. This action preferentially exhausts any airborne radioactive products through the filtered SGTS and out the main stack in order to minimize offsite releases.

   The design basis event of interest with regard to damper timing is the refueling accident and is described in Chapter 14 of the FSAR. An additional discussion of the system design basis was also included in the

FSAR, section 5.3.4.2, and is repeated in the following excerpt for reference.* There are no technical specification requirements for response time testing of the dampers.

FSAR Section 5.3.4.2

The total time required to switch from the normal containment ventilation system to the SGTS upon detection of high radiation is relatively small. The radiation monitor response time is one second, and the reactor zone isolation dampers (pneumatic drive) are closed in two seconds upon receipt of signals from the monitor. Startup of the SGTS blower motors is within five seconds from the time of the signal. The proper dampers (electric motor-driven) in the SGTS trains are fully opened in 30 seconds. The one-second response on radiation detection, plus the two-second response on the isolation damper, is less than the transport time from the surface of the fuel pool to the isolation damper, and thereby assures that the release from a fuel handling accident will be contained in the secondary containment. All SGTS trains will be producing full flow within 30 seconds of a secondary containment isolation signal.

Important points from the preceding description are:

a. The exhaust dampers that are assumed to close in two seconds to limit offsite releases are apparently the refuel zone dampers.

b. Original analysis assumed that damper closure time was short compared to the transport time of airborne fission products and, therefore, that the fission product release via the ventilation system was zero.

c. The description does not make a specific reference to the refuel zone exhaust dampers with regard to assumed closure times. Only reactor zone dampers are addressed.

In preparation for a changeout of the air solenoids on these exhaust dampers for the environmental qualification program, a request for post-modification test criteria for damper closing times was directed to the engineer for secondary containment. The cognizant engineer reviewed the FSAR basis, preoperational test results, procurement specifications, and also tested the actual damper closing times. The test results of September 23, 1985 showed that refuel zone exhaust and supply dampers met the two second criteria except for two of the 12 dampers. Minor adjustments to the pressure regulator were made, and the refuel zone exhaust damper came back within the two seconds. The reactor zone dampers averaged about three seconds with the high value being five seconds and could not be appreciably speeded up. The reactor zone dampers are physically larger than the refuel zone dampers which basically accounts for the closure time difference.

*(This excerpt is as the FSAR read during the period of the TVA evaluation.)

These findings were weighed against the FSAR description and Chapter 14 analysis of the fuel handling accident. It was obvious that the speed of the reactor zone dampers would have very little effect on the accident analysis. This is due to the geometric separation of the respective zones and ductwork, coupled with the fact that the refuel zone ductwork is physically arranged such that any releases from fuel handling accidents are preferentially exhausted through the refueling zone exhaust ducting. The plant staff concluded that the FSAR wording was incorrect with regard to the damper nomenclature. From an engineering and safety viewpoint, the closure time of the reactor zone dampers posed no problem. Nevertheless, as a conservative action, an administrative restriction on fuel handling was imposed on September 23, 1985 while the test results were being further investigated. The Resident Inspectors were notified of our findings and kept informed concerning our activities to address the subject. The Residents did express their concerns relative to our decision not to declare secondary containment inoperable and carried the issue as an unresolved item in Inspection Report 85-49 (November 7, 1985). The Plant Engineering Staff had some uncertainty in justifying the assumptions of the original Chapter 14 analysis concerning closure time of the refueling zone exhaust dampers compared to the physical transport time of fission gases through the ductwork. This is because the analysis did not appear to include any ground level release term, which means that no gas escapes through the refuel zone ductwork. This assumption seemed to be nonconservative.

To fully resolve the issue, TVA decided to contract GE to perform an entirely new refueling accident analysis and safety evaluation. This analysis was expected to confirm our conclusions relative to the importance of the damper closing times and to provide a modern analysis for reference. The new analysis utilized Regulatory Guide 1.25 and NUREG-0800 (Standard Review Plan) guidance. We also included in the contract a provision for a sensitivity study assuming two-, five-, and 10-second damper closure times to gain insight into the effect of this parameter on calculated offsite doses from a fuel handling accident.

The new fuel handling accident analysis was completed on November 7, 1985 and the calculated results for the three selected closure times showed that predicted offsite releases were a small fraction of 10 CFR 100 criteria. The analysis also confirmed that the reactor zone damper timing did not affect the analysis results and that the release dose magnitude varied directly with the timing of the refuel zone exhaust dampers. A decision was made to clarify the FSAR during the annual update to correctly reference the refuel zone damper function (which was done by our recent FSAR update). An internal plant memorandum was written to the Operations Supervisor summarizing the analysis conclusions, and the administrative restrictions on fuel handling were subsequently lifted in late November 1985. The subject memorandum did not authorize changes to be made in the closure speeds of the damper, but rather confirmed that the as-found speeds were acceptable.

The Resident Inspectors reviewed the GE analysis and suggested that the use of the analysis results to justify longer stroke times was a potential USQ as defined in 10 CFR 50.59. They also inquired concerning why a 10 CFR 50.59 evaluation had not been performed. The Plant Engineering Staff was of the opinion that the situation was not within the scope of 10 CFR 50.59, since no actual plant changes or procedure changes were necessary. Furthermore, it had been previously demonstrated that the refuel zone exhaust dampers could meet the existing FSAR description, and the new analysis confirmed that they were the only dampers affecting the offsite release calculations. Closure of the issue was planned with a revision of the FSAR during the annual update, at which time a formal 10 CFR 50.59 evaluation would be performed. (This year's annual FSAR update did subsequently incorporate the new fuel handling accident analysis.) To address the Inspector's concern, the Plant Manager requested his staff to perform a 10 CFR 50.59 evaluation on the new analysis. The Plant Engineering Staff prepared a 10 CFR 50.59 evaluation and concluded that utilization of the new analysis results would not constitute a USQ. The Plant Operations Review Committee reviewed and concurred with the evaluation. On December 13, 1985 the Resident Inspector notified the Plant Manager that they disagreed with the 10 CFR 50.59 evaluation, as was later documented in Inspection Report 85-57 as a potential escalated enforcement item. The Plant Manager reimposed limited administrative fuel handling restrictions pending resolution of the impasse.

TVA sought to resolve the disagreement by formally requesting a Nuclear Reactor Regulation (NRR) review and concurrence with the 10 CFR 50.59 evaluations. The submittal letter was mailed on January 23, 1986 and requested NRR to review the new fuel handling accident analysis and to concur with the plant staff's conclusions and 10 CFR 50.59 evaluation. On March 28, 1986 NRR responded and agreed with the TVA evaluation. Fuel handling restrictions were then removed by the Plant Manager.

This item was discussed in the May 28, 1986 enforcement conference and was subsequently categorized as a Level IV violation as a failure to perform a 10 CFR 50.59 evaluation on a facility change.

In summary, we do not believe a violation of 10 CFR 50.59 occurred since there was not a change to the facility or its procedures. Also, as can be seen by the chronology, we treated the situation in an extremely conservative manner in both applying voluntary fuel handling restrictions, as well as purchasing a modern analysis of the fuel handling analysis. We also responded to the Resident Inspector's concerns in a timely manner. Accordingly, we request that NRC reconsider this proposed violation.