August 31, 20004

LICENSEE: Tennessee Valley Authority

FACILITY: Browns Ferry Nuclear Plant, Units 1, 2 and 3

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JULY 12, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION (NRC) AND THE TENNESSEE VALLEY AUTHORITY (TVA) CONCERNING REQUESTS FOR ADDITIONAL INFORMATION (RAI) ON BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. MC1704, MC1705 AND MC1706)

The U.S. Nuclear Regulatory Commission staff and representatives of Tennessee Valley Authority (TVA or the applicant) held a telephone conference call on July 12, 2004, to discuss the requests for additional information (RAIs) related to Section 3.5, Unit 1 structural components during extended outage (Unit 1 lay-up condition) of the Browns Ferry Nuclear Plant (BFN) license renewal application.

The conference call was useful in clarifying both the staff’s questions and the applicant’s responses to those questions. On the basis of the discussions, the applicant was able to better understand the intent of the staff's RAIs. No staff decisions were made during the meeting.

Enclosure 1 provides a listing of the RAIs discussed with the applicant, including a brief description on the status of the items. Enclosure 2 contains a list of the telephone conference call participants. The applicant has provided formal responses to the RAIs via letter dated on July 19, 2004.

/RA/

Yoira K. Diaz Sanabria, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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

Enclosures: As stated

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August 31, 2004

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SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JULY 12, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION (NRC) AND THE TENNESSEE VALLEY AUTHORITY (TVA) CONCERNING REQUESTS FOR ADDITIONAL INFORMATION (RAI) ON BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3, LICENSE RENEWAL APPLICATION (TAC NOS. MC1704, MC1705 AND MC1706)

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BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)
RELATED TO THE LAY-UP EFFECTS OF BROWNS FERRY UNIT 1
STRUCTURES AND COMPONENTS SUPPORTS

RAI 3.5-1

Browns Ferry Nuclear Plant (BFN) document titled, "Evaluation of the BFN Unit 1 Lay-Up and Preservation Program," including Tables 1 through 4, does not provide information related to BFN's evaluation of the Unit 1 spent fuel storage system lay-up effects. Please describe the method adopted in assessing the Unit 1 spent fuel storage system related lay-up effects. Also, provide a discussion of the applicable spent fuel pool environments (any delta change in pool water chemistry, ambient humidity and temperature, etc.), results of past periodic inspections of the spent fuel pool structural components and pool liners, any observed pool leakages or degraded conditions, and corrective actions taken to support BFN's conclusion that no lay-up effect is applicable to Unit 1 spent fuel storage system.

TVA response:

The Unit 1 spent fuel storage system was never placed in lay-up. The Unit 1 spent fuel storage system contains spent fuel and remained in service since Unit 1 was shut down and defueled in 1985. The Unit 1 spent fuel storage pool is located on elevation 664.0' of the Unit 1 reactor building. This area where the spent fuel pools are located is referred to as the refuel floor and is common for all three units (i.e., there are no physical barriers separating the spent fuel pools from the other units). Therefore the spent fuel pools are exposed to the same operating environments. The spent fuel storage pool chemistry is maintained in accordance with Technical Requirement Manual Section TR 3.9.3 Spent Fuel Pool Water Chemistry.

The spent fuel pool storage system is in-service and complies with all applicable license and regulatory requirements. The structural components of the Unit 1 spent fuel storage system are being monitored under the Maintenance Rule (Structures Monitoring Program [SMP]) requirements, which are the same requirements for inspection of Units 2 and 3 spent fuel storage system. Plant procedure 0-TI-346 implements the requirements of the Maintenance Rule and contains the same performance criteria for all 3 units. The Maintenance Rule inspection results for Unit 1 spent fuel storage pool are consistent with the Maintenance Rule inspection results for Units 2 and 3 spent fuel storage pools. The structural components of the Unit 1 spent fuel pool and the supporting equipment of the spent fuel pool storage system are all exposed to an environment that is consistent with the operating environments of the Units 2 and 3 spent fuel storage system. Any degraded condition discovered during system operation or as part of the Maintenance Rule inspection of the Unit 1 spent fuel storage system is handled the same as for the Units 2 and 3 spent fuel storage systems. The Browns Ferry corrective action program to address degraded conditions is SPP-3.1. The structural components of the Units 1, 2 and 3 spent fuel storage system are addressed in LRA Section 2.4.2.1.

The operating environment for the Unit 1 spent fuel storage system is consistent with the operating environments of the Units 2 and 3 spent fuel storage systems and the system has been maintained consistent with license and regulatory requirements and plant corrective program. Therefore there is no difference in the Unit 1 spent fuel storage system from Units 2 and 3. Since the system was not in lay-up as described above, no lay-up effects are applicable to the Unit 1 system. This is the basis for not including the spent fuel storage system to the BFN document "Evaluation of the BFN Unit 1 Lay-Up and Preservation Program."

Discussion: The applicant stated that it would review their response to determine if any changes needed to be made based on the staff's comment.

RAI 3.5-2

Please describe the approach used in evaluating the Unit 1 structures and component supports related lay-up effects. Provide a discussion of the environments applicable to Unit 1 structures and component supports (e.g., any exposure to aggressive chemicals or ponding of water, significant change in ambient humidity and temperature, etc.), results of past periodic inspections of the structures and component supports, any observed degraded conditions, and corrective actions taken to support BFN's conclusion that no lay-up effect is applicable to Unit 1 structures and component supports that require an Aging Management Review (AMR).

TVA response:

For Unit 1 structures and component supports, the external service environments defined in Table 3.0.2 of the LRA were used in the aging management review. An example of an environment is the Inside Air environment that is defined in Table 3.0.2 as "Atmospheric air, maximum average temperature 150^{o}F, humidity up to 100%, potentially exposed to ionizing radiation, not exposed to weather." The range of interior temperatures, pressures, relative humidity and radiation dose for the reactor building and primary containment are defined in calculations ND-Q1999-900031 (RIMS W78 030430 005), "Summary of Operational Environmental Conditions for Browns Ferry Nuclear Plant," ND-Q2999-880143 (RIMS R14 020723 105), "Summary of Harsh Environmental Conditions for Browns Ferry Unit 2" and ND-Q3999-910035 (RIMS R14 020723 104), "Summary of Harsh Environmental Conditions for Browns Ferry Unit 3." The interior temperatures, pressures, relative humidity and radiation dose are shown on the Harsh Environmental Data Drawings 47E225 series for each unit. The environmental conditions defined in the referenced calculations are enveloped by the definition for "Inside Air" contained in Table 3.0.2, except for the area of the main steam tunnel located on elevation 565.0' of the Units 2 & 3 reactor buildings. The main steam tunnels during plant operation have an average area temperature of 160^{o}F. This temperature occurs as a result of plant operation and has not been seen in the same area of the Unit 1 reactor building during plant lay-up. The Unit 1 lay-up environment is the same or bounded by the evaluated operating environments.

The Unit 1 reactor building structure is subject to the Maintenance Rule (Structures Monitoring Program [SMP]) requirements. A baseline inspection for the Browns Ferry (BFN) SMP was performed in 1997. All the same attribute inspections that were performed for Units 2 and 3 were performed for Unit 1. This inspection is documented in calculation CD-Q0303-970086 (RIMS R14 971105 102). LCEI-CI-C9, "Procedure for Walkdown of Structures for Maintenance Rule," was the procedure utilized to perform SMP inspections and requires the documentation of defects in accordance with the requirements of the procedure. There were two defects noted from the inspection of the Unit 1 reactor building and these two defects were noted as: (1) a personnel lock door that appeared to not be air tight and (2) rust was noted on some of the torus reinforcement steel between bays 12-13, 13-14 and 14-15. These defects were dispositioned as not affecting the function of the structural component. The SMP requires a reinspection on a five year frequency. The 2002 SMP inspection is documented in calculation CDQ0-303-2003-0260 (RIMS R14 030211 102). During the 2002 SMP inspections, there were four defects noted from the inspection of the Unit 1 reactor building and were dispositioned as not affecting the function of the structural component. These four defects were noted as: (1) a concrete pad at the floor around a conduit was chipped, (2) bolt missing from angle securing structural plate partition wall to the concrete floor (3) in the south west corner of the stairwell between elevation 593' and 621', mortar was missing at one end of the masonry block and (4) some concrete deterioration was noted in bay 7 of the torus area (work was in progress to repair the area and was noted in the walkdown). These defects noted from the two inspection periods can be categorized as isolated conditions and do not represent an adverse trend that will affect the functionally of structural components.

The component supports located in Unit 1, except for those that are required for Units 2 or 3 system operation, are not subject to periodic inspections during the shut down period. All component supports for safety-related systems required for Unit 1 operation were inspected and existing configuration confirmed as part of the Unit 1 recovery effort. The following plant procedures [walkdown instructions (WI)], were utilized: WI-BFN-0-CEB-01 was used for piping and supports, WI-BFN-0-CEB-02 was used for structural items, and WI-BFN-0-GEN-01 was used for both piping/supports and structural steel as a general walkdown procedure. Additionally, the following procedures were used to document baseline configuration for other component supports:

WI-BFN-0-CEB-03 - Small Bore Piping
WI-BFN-0-CEB-04 - Seismic Verification of A46 and IPEEE
WI-BFN-0-CEB-05 - Pipe Rupture/HELB
WI-BFN-0-CEB-06 - Seismically Induced Water Spray

The inspections document as-built configuration or existing plant configurations that were not in conformance with the acceptance criteria defined in the WI. These configurations were evaluated to design criteria requirements. If the evaluations determined that the configuration does not meet the design criteria requirement, a plant modification was designed and issued under plant work control process.

An electronic search of the site corrective action program for PERs was performed to identify any adverse conditions with component supports. The search did not result in the identification of any adverse conditions.

The environment for the Unit 1 structures and component supports is consistent with the operating environments of the Units 2 and 3 structures and component supports, therefore there is no difference in the Unit 1 structures and component supports from Units 2 and 3 and no lay-up effects are applicable to Unit 1.

Discussion: The applicant stated that it would review their response to determine if any changes needed to be made based on the staff's comment.

RAI 3.5-3

When the plant is operating, the containment drywell, torus, and connecting vent assemblies are subjected to relatively inert environment, and all the requirements related to their inspections, and leak-rate testing are applicable. These requirements ensure the leak tight, and structural integrity of these components. Also, the industry operating experience problems, as reflected in NRC's Generic Letters, Information Notices, and other industry published event reports are considered as applicable. These activities may or may not have been considered for the Unit 1 during its long lay-up. In this context, the applicant is requested to provide information that would describe the benchmark condition of the containment pressure boundary related components prior to restart of the Unit, and actions that will be taken prior to the extended period of operation. The relevant regulatory requirements would be: 10 CFR 50.55a, and Appendix J of 10 CFR 50. The relevant Generic Letters would be: GL 87-05, GL 89-16, and GL 98-05. The relevant Information Notices would be: IN 86-99, IN 88-82, IN 89-06, IN 89-79, and IN 92-20.

TVA Response:

For the Unit 1 containment drywell and torus, the environment during the extended outage was the same or bounded by the evaluated operating unit environments.

LRA Table 3.0.2 describes the containment environment for the drywell and torus that was used in the aging management review as:

"Atmospheric air, maximum average temperature 150°F, humidity up to 100%, potentially exposed to ionizing radiation, not exposed to weather."

Inerting was not credited for elimination of aging effects requiring aging management.
Also note that the Unit 1 containment environment associated with temperature and ionizing radiation are not as severe as the evaluated (operating) environment conditions.

The torus was subject to the torus water environment during the shutdown period. The torus has subsequently been drained and is being refurbished as part of Unit 1 recovery effort.

Containment inspections, and leak-rate testing 100% of the examinations required in Examination Categories of Table IWE-2500-1 for the First Inspection Interval will be completed as preservice exams before Unit 1 restarts except those that may be excluded by 10 CFR 50.55a and except where specific written relief has been granted by the NRC. The requirements of ASME Section XI In-Service Inspection Subsection IWE, 1992 Edition with the 1992 Addenda will be implemented on Unit 1. Type A, B & C leak rate testing required by 10 CFR 50 Appendix J will also be performed prior to Unit 1 restart.

Consideration of NRC Generic Communications

GL 87-05

NRC Generic Letter 87-05: Request Additional Information Assessment - Degradation of Mark I Drywells TVA provided the NRC with the results of the ultrasonic testing for corrosion degradation of the drywell liner plate, RIMs No. L44 880830 801, dated August 30, 1988. The results of the ultrasonic testing states: Each unit's drywell was ultrasonically tested near the sand cushion area during 1987. The results from these tests showed that the nominal thickness was maintained on each drywell. On Unit 1, no reading below the nominal thickness of one inch was measured indicating that the integrity of the drywell liner plate was maintained.

GL 89-16

NRC Generic Letter 89-16: Installation of a Hardened Wet Well Vent
Browns Ferry will be installing the hardened well vent as part of the Unit 1 recovery effort. This generic letter does not address aging effects or aging management considerations.

GL 98-05

NRC Generic Letter 98-05: Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Vessel Circumferential Shell Welds

This Generic Letter is not applicable to the containment drywell, torus, and connecting vent assemblies.

IN 86-99

NRC Information Notice 86-99: Degradation of Steel Containments
See response to Generic letter 87-05

IN 88-82

NRC Information Notice 88-82: Torus Shells with Corrosion and Degraded Coatings on BWR Containments

In 1983, Engineering Change Notice (ECN) P0555 was issued to completely inspect and recoat the tori as necessary. The Unit 1 work was completed on this ECN:

IN 89-06

NRC Information Notice 89-06 Bent Anchor Bolts in Boiling Water Reactor Torus Supports

Based on the configuration of the Browns Ferry torus supports, it has been determined that BFN tie down bolts would not be subject to the effects that occurred at plant Hatch. This information notice does not address aging effects or aging management considerations.

IN 89-79

NRC Information Notice 89-79: Degraded Coatings and Corrosion of Steel Containment Vessels

This information notice addresses corrosion of steel ice condenser containments. Corrosion of the BFN containment drywell, torus, and connecting vent assemblies was addressed as indicated in GL 87-05 and IN 88-82

IN 92-20

NRC Information Notice 92-20: Inadequate Local Leak Rate Testing

The vent line bellows at Browns Ferry are a different design (single ply bellows) than the Quad Cities bellows identified in IN 92-20. The design of the Browns Ferry penetration bellows allows full pressure to be transmitted to all portions of the bellows during Appendix J testing.

Discussion: The staff indicated that in generic letter (GL 89-16) and in information notice (IN 89-06) the statement, "This generic letter/information notice does not address aging effects or aging management considerations" is not appropriate. The applicant stated that it would review their response to determine if any changes needed to be made based on the staff's comment.

LIST OF PARTICIPANTS FOR TELEPHONE CONFERENCE ON REQUESTS FOR ADDITIONAL INFORMATION

JULY 12, 2004

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Enclosure 2

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