



Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

July 19, 2004

10 CFR 54
10 CFR 51

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of)	Docket Nos. 50-259
Tennessee Valley Authority)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN), UNITS 1, 2 AND 3 (TAC NOS. MC1768, MC 1769, AND MC1770) LICENSE RENEWAL APPLICATION: RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING LAY-UP EFFECTS OF UNIT 1 STRUCTURES AND COMPONENT SUPPORTS

By letter dated June 23, 2004, the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the Tennessee Valley Authority's (TVA's) analysis of lay-up effects of Unit 1 structures and component supports submitted with TVA's application to renew the operating licenses for BFN, Units 1, 2, and 3. Enclosed is TVA's response to the NRC staff's request for additional information.

This letter contains no new commitments.

If you have any questions about this information, please contact Gary Adkins, Manager, BFN License Renewal Project, at (423) 751-4363 or gmadkins@tva.gov.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this nineteenth day of July 2004.

Sincerely,


 Mark J. Burzynski
 Manager
 Nuclear Licensing

D030

U.S. Nuclear Regulatory Commission
Page 2
July 19, 2004

Enclosure

cc: State Health Officer
Alabama Department of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, Alabama 36130-3017

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. Stephen J. Cahill, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

NRC Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

NRC Unit 1 Restart Senior Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611-6970

Mr. Kahtan N. Jabbour, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 08G9
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Ms. Eva A. Brown, Project Manager
U. S. Nuclear Regulatory Commission
MS 08G9
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2738

cc: Continued on page 3

U.S. Nuclear Regulatory Commission
Page 3
July 19, 2004

cc: Mr. William F. Burton, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 011F1
Two White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2738

Yoira K. Diaz-Sanabria, Project Manager (w/enclosure)
U.S. Nuclear Regulatory Commission
MS 011-F1
Two White Flint, North
11545 Rockville Pike
Rockville, Maryland 20852-2738

ENCLOSURE

BROWNS FERRY NUCLEAR PLANT (BFN), UNITS 1, 2 AND 3, LICENSE RENEWAL APPLICATION: TVA RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION (RAIs) RELATED TO THE LAY-UP EFFECTS OF UNIT 1 STRUCTURES AND COMPONENTS

RAI 3.5-1

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 to the same criteria for all three units, in accordance with Technical Requirements Manual section TR 3.9.3, Spent Fuel Pool Water Chemistry.

The spent fuel pool storage system is an in-service system and governed by 10 CFR regulations, including the requirements of the Maintenance Rule. 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 the Units 2 and 3 spent fuel storage system. Plant procedure 0-TI-346 implements the requirement of the Maintenance Rule. 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 that would be discovered during system operation or as part of the Maintenance Rule inspections of the Unit 1 spent fuel storage system would be handled the same as for the Units 2 and 3 spent fuel storage systems. The BFN site 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 requirements 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; therefore, there is no difference in the Unit 1 spent fuel storage system from Units 2 and 3 and no lay-up affects 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."

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°F, humidity up to 100 percent, 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 and 3 reactor buildings. The main steam tunnels during plant operation have an average area temperature of 160°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 SMP requirements. A baseline inspection for the 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 airtight 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 structural function. 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 which were dispositioned as not affecting structural function. These four defects were noted as: (1) a concrete pad at the floor around conduit was chipped, (2) bolt missing from angle securing

structural plate partition wall to the concrete floor, (3) in the southwest corner of a stairwell, 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 in progress to repair the area was noted from 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 functionality of structural components.

The component supports located in Unit 1, except for those that are required for Unit 2 or Unit 3 system operation, are not subject to periodic inspections during the shutdown period. All component supports for safety-related systems required for Unit 1 operation are to be inspected and existing configurations confirmed as part of the Unit 1 recovery effort. The following plant procedures (walkdown instructions [WI]), are 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 configurations 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 would document as-built configurations or existing plant configurations that did not conform to the acceptance criteria defined in the WI. These configurations would be evaluated to design criteria requirements. If the evaluations determined that the configuration did not meet the design criteria requirement, a plant modification would be designed and issued under the 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 affects are applicable to Unit 1.

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 (GL), Information Notices, and other industry-published event reports are considered as applicable. These activities may or may not have been considered for 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 GLs would be: GL 87-05, GL 89-16, and GL 98-05. The relevant Information Notices (IN) 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 percent, 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, which is the same environment as experienced by Units 2 and 3. The torus has subsequently been drained and is being refurbished as part of the Unit 1 recovery effort.

Containment Inspections, And Leak-Rate Testing

One hundred percent 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 and C leak rate testing required by 10CFR50, Appendix J, will also be performed prior to Unit 1 restart.

Consideration of NRC Generic Communications

GL 87-05

NRC GL 87-05: Request Additional Information Assessment - Degradation of Mark I Drywells
TVA provided NRC with the results of the ultrasonic testing for corrosion degradation of the drywell liner plate, RIMS No. L44 880830 801, dated Aug. 30, 1988. The results of the ultrasonic testing state: 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 GL 89-16: Installation of a Hardened Wet Well Vent BFN will be installing the hardened well vent as part of the Unit 1 recovery effort.

GL 98-05

NRC GL 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 GL is not applicable to the containment drywell, torus, and connecting vent assemblies.

IN 86-99

NRC IN 86-99: Degradation of Steel Containments
See response to GL 87-05.

IN 88-82

NRC IN 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 IN 89-06: Bent Anchor Bolts in Boiling Water Reactor Torus Supports
Based on the configuration of the BFN torus supports, it has been determined that BFN tie down bolts would not be subject to the effects that occurred at plant Hatch.

IN 89-79

NRC IN 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 IN 92-20: Inadequate Local Leak Rate Testing
The vent line bellows at BFN are a different design (single ply bellows) than the Quad Cities bellows identified in IN 92-20. The design of the BFN penetration bellows allows full pressure to be transmitted to all portions of the bellows during Appendix J testing.