11-16-78

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

Mr. N. B. Hughes
Manager of Power
Tennessee Valley Authority
830 Power Building
Chattanooga, Tennessee 37401

44
40
17

Dear Mr. Hughes:

The Commission has issued the enclosed Amendments Nos. , and to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your requests of August 2, 1978 (BFNP TS 112) and August 11, 1978 (BFNP TS 114).

The changes: (1) permit the average power range monitor system to be inoperable in the refuel mode, provided the source range monitors are connected to give a non-coincidence, high flux scram; (2) permit less than three intermediate range monitors per trip channel to be operable in the shutdown or refuel modes, provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram; (3) clarify ambiguous portions of the Technical Specifications related to the rod block monitor system; (4) remove reference to an obsolete 1968 version of an ASTM procedure; (5) modify the list of snubbers that are required to be operable; (6) remove a specification for additional tests of secondary containment that only applied during the first fuel cycle for each Browns Ferry Unit; and (7) alter one of the four locations where milk samples are collected. With the concurrence of your staff, we have made several minor changes in the proposed Technical Specifications which you submitted.

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cep

OFFICE ▶						
SURNAME ▶						
DATE ▶						

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. to DPR-33
2. Amendment No. to DPR-52
3. Amendment No. to DPR-68
4. Safety Evaluation
5. Notice

} See Rpts
} w/ this Ltr

cc w/enclosures:
see next page

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DOCKETS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 16, 1978

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

Mr. N. B. Hughes
Manager of Power
Tennessee Valley Authority
830 Power Building
Chattanooga, Tennessee 37401

Dear Mr. Hughes:

The Commission has issued the enclosed Amendments Nos. 44, 40 and 17 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your requests of August 2, 1978 (BFNP TS 112) and August 11, 1978 (BFNP TS 114).

The changes: (1) permit the average power range monitor system to be inoperable in the refuel mode, provided the source range monitors are connected to give a non-coincidence, high flux scram; (2) permit less than three intermediate range monitors per trip channel to be operable in the shutdown or refuel modes, provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram; (3) clarify ambiguous portions of the Technical Specifications related to the rod block monitor system; (4) remove reference to an obsolete 1968 version of an ASTM procedure; (5) modify the list of snubbers that are required to be operable; (6) remove a specification for additional tests of secondary containment that only applied during the first fuel cycle for each Browns Ferry Unit, and (7) alter one of the four locations where milk samples are collected. With the concurrence of your staff, we have made several minor changes in the proposed Technical Specifications which you submitted.

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Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 44 to DPR-33
2. Amendment No. 40 to DPR-52
3. Amendment No. 17 to DPR-68

} See Rpts

4. Safety Evaluation
5. Notice

} w/THIS LTR.

cc w/enclosures:
see next page

Tennessee Valley Authority - -

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS NOS. 1, 2 AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 Introduction

By letter dated August 11, 1978 (TVA BFNP TS 114), the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would (1) permit the average power range monitor (APRM) system to be inoperable in the refuel mode, provided the source range monitors (SRMs) are connected to give a non-coincidence, high flux scram and (2) in the refuel and shutdown modes only, permit less than three intermediate range monitors (IRMs) per trip channel to be operable-provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram. The present Technical Specifications require that a minimum of three IRMs per trip channel be operable at all times (i.e., shutdown as well as startup and operation).

The reason for this request is to allow the interchange of the fission chambers in the current APRM system with reduced radiation exposure to the operating personnel and with reduced handling and movement of fuel. This can be achieved by removing many LPRMs simmultaneously rather than in sequence. The sequential removal would leave the APRM system operable but the simultaneous removal would not.

In a separate letter dated August 2, 1978 (TVA BFNP TS 112), TVA requested five changes to the Technical Specifications, all of which are administrative in nature. The changes would: (1) clarify an ambiguious portion of the Technical Specifications related to the rod block monitor system, (2) remove reference to an obsolete 1968 version of an ASTM procedure, (3) modify the list of snubbers that are required to be operable, (4) change one of the four locations from which milk samples are routinely collected and (5) remove a specification for additional test of secondary containment that only applied to the first operating cycle for each Browns Ferry unit.

2.0 Discussion

As described in Section 7.5 of the Final Safety Analysis Report (FSAR) for the Browns Ferry Nuclear Plant (BFNP), the Neutron Monitoring System consists of six major subsystems: (a) the Source Range Monitor (SRM) subsystem, (b) the Intermediate Range Monitor (IRM) subsystem, (c) the Local Power Range Monitor (LPRM) subsystem, (d) the Average Power Range Monitor (APRM) subsystem, (e) the Rod Block Monitor (RBM) subsystem and (f) the Traversing In-Core Probe (TIP) subsystem. The IRM subsystem monitors neutron flux from the upper portion of the SRM range to the lower portion of the Power Range Monitoring Subsystems.

The IRM system normally consists of eight moveable miniature chambers with two such chambers in each core quadrant. No more than one of the IRMs in each quadrant may be bypassed. The eight IRM channels are divided into two IRM sub-systems and at least one IRM from each sub-system must reach 120/125 of full scale to initiate a reactor scram. The IRM system is nominally designed for protection in the startup mode and analyses (FSAR, Section 14.5.3) have been performed showing that the system adequately prevents fuel damage due to rod withdrawal errors postulated to occur during startup.

The APRM subsystem provides a continuous indication of average reactor power from a few percent to 125% of rated reactor power. The subsystem has six APRM channels, each of which uses input signals from a number of LPRM channels. Three APRM channels are associated with each of the trip systems of the Reactor Protection System.

The APRM system which consists of a number of stationary fission chambers dispersed throughout the core, is normally required to be operable in the refuel mode with a high flux scram setpoint corresponding to 15% rated power.

Because the APRM response is actually the combined response of a number of individual fission chambers located throughout the core, the APRM primarily provides protection for core-wide transient power increases which might occur in the run mode (above 15% rated power). Also, in the startup mode the APRM provides backup protection to the IRM system against localized power increases which might result from postulated rod withdrawal errors.

Although the IRM system as described above is required by the current Technical Specifications to be operable in both the shutdown and refuel modes, no specific event has been analyzed in the Plant FSAR which takes credit for a scram initiated by the IRM system with a given setpoint or number of bypassed instruments. Similarly, the APRM is required to operate normally in the refuel mode, but no transient or accident taking credit for an APRM initiated scram, and postulated to occur in the refuel mode has been analyzed in the Plant FSAR. As discussed in the evaluation which follows, there is only one event which the staff can postulate - namely, an operator bypassing the interlocks and withdrawing a second control rod adjacent to one which is already withdrawn - for which the IRM/APRM subsystems are required to provide safety protection in the refuel and shutdown modes.

Section 14.5.3 of the Browns Ferry FSAR discusses the events that could result directly in positive reactivity insertions, including control rod removal error during refueling and fuel assembly insertion error during refueling. Section 7.6 of the FSAR describes the refueling interlocks that prevent an inadvertent criticality during refueling operations and that are designed to back up procedural core reactivity controls during refueling operations. Section 3.10 of the Browns Ferry Nuclear Plant Technical Specifications lists the restrictions that apply during core alterations to ensure that core reactivity is within the capability of the control rods and to prevent criticality during refueling.

When the mode switch is in REFUEL, only one control rod can be withdrawn. Selection of a second rod initiates a rod block thereby preventing the withdrawal of more than one rod at a time. The Refueling Interlocks, in combination with core nuclear design and refueling procedures, prevent inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. Refueling procedures are written to avoid situations in which inadvertent criticality is possible. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations when the mode switch is in REFUEL position. The interlocks on hoists provide yet another method of avoiding inadvertent

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The present Technical Specifications permit bypassing the refueling interlock with the requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed to ensure that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shut down by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38% Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

3.0 Evaluation

3.1 APRM-IRM Systems

We have reviewed the plant Technical Specifications and the nuclear design characteristics of the fuel. We have concluded that a local criticality during shutdown or refueling operations could only occur through violation of technical specifications such as an operator error in withdrawing a control rod for maintenance, adjacent to a previously withdrawn rod.

Although such operator errors are not likely to occur, they are not impossible. We have therefore considered the applicant's request for proposed modifications to the SRM, IRM and APRM systems in terms of the impact on the protection against postulated local criticality which could occur while the mode selection switch is in the refuel or shutdown positions.

The most severe test of the adequacy of the modified IRM and SRM systems would be the withdrawal (for maintenance) of a control rod near the edge of the reactor core face adjacent to a previously withdrawn rod. Because the proposed Technical Specifications allow one IRM in each core quadrant to be bypassed, the IRM nearest the pair of withdrawn rods was assumed to be bypassed.

Because the modified IRM system would initiate a reactor scram when any IRM reaches the trip set point, the modified system will actuate a scram at an earlier time during the withdrawal of the second rod than would the normal system. The normal system would require trips in each IRM subsystem.

We conclude that the redundant independent IRM instruments connected to give non-coincident scrams provide better protection against fuel damage due to a localized power increase than does the APRM system with its 15% scram setpoint. Beacuse the IRM instruments are independent in the modified IRM system, the IRM will be its own backup. The IRM scram setpoint will be 120/125 of the lowest IRM scale which corresponds to very low flux levels. Although the flux level at the second nearest IRM (the backup IRM) would be low throughout the rod withdrawal event, it will be high enough to scram the reactor at a lower flux level than with the present arrangement using the APRM monitors. We therefore, conclude that the licensee's proposal for the IRM system modification results in a system that is more sensitive to possible operator errors during core modifications than is the present arrangement and therefore the proposed modification is acceptable.

In addition, the SRM system would be connected to scram the reactor at a level of 5×10^5 counts per second. Although the SRM is not considered safety grade equipment, the licensee has proposed to provide the SRM scram function, and we believe this is desirable as an additional backup to the IRM system.

A concern which was raised during the NRC review was what technique(s) will be provided to assure that the reconfiguration of the SRM's and IRM's to the non-coincidence trip mode is in fact accomplished prior to removing the APRM protection. By letter dated November 13, 1978, the licensee has agreed to the following administrative controls. The procedures related to maintenance of detectors ("Browns Ferry Nuclear Plant-Instrument Maintenance Instructions") will be reviewed, and revised as necessary, to include: (1) a specific reference to the Technical Specification Table 3.1.A and associated Notes 21 and 22, which indicate that the SRM's/IRM's must be re-configured to provide non-coincidence high flux scram protection, and (2) a specific procedural step which requires that verification will be made that the appropriate shorting links have been removed prior to maintenance on IRM/LPRM detectors. These controls provide adequate assurance that the reconfiguration of the SRMs and IRMs will be accomplished prior to removing the APRM protection.

Due to the interwoven design of the shorting link system, clarification of the notes to Table 3.1.A is needed. The following sentence should be added to Note 21: "The removal of eight (8) shorting links is required to provide non-coincidence high-flux scram protection from the Source Range Monitors". The following sentence should be added to Note 22: "The removal of four (4) shorting links is required to provide non-coincidence high-flux scram protection from the IRM's".

As is proposed by the licensee for Unit No. 3, the Technical Specifications for Units Nos. 1 and 2 should include in Note 21 to Table 3.1.A that the scram setpoint is $\leq 5 \times 10^5$ CPS.

To summarize, we find that the modification TVA has proposed for the Browns Ferry IRM systems is acceptable. The modified systems will be more sensitive to the flux perturbations resulting from the worst postulated transient than the present arrangement. Furthermore, as discussed previously, the redundant and independent IRM instruments which will comprise the modified IRM systems will provide protection against inadvertent criticality in the refuel mode equivalent to or better than the present APRM system. Inoperability of the APRM with the modified IRM in place is therefore acceptable for the refuel mode.

As described in the "Discussion" above, Section 3.10 of the Technical Specifications includes restrictions on withdrawal of control rods during core alterations. As an additional backup to the neutron monitoring instrumentation, we have proposed, and the licensee has accepted, an addition to the surveillance requirements in Section 4.10 of the Technical Specifications to require that no control rod may be withdrawn for maintenance until two licensed operators have confirmed that there is no fuel in the cell controlled by the particular control rod or that all immediately adjacent control rods are fully inserted and electrically disarmed. This requirement, in conjunction with the more sensitive IRM system, will insure that there is no possibility of inadvertent criticality during core modifications.

In summary we conclude that the proposed changes to the licensee's Technical Specifications do not involve an increase in the probability of a transient or accident but in fact should reduce the consequences of such events. The proposed changes do not involve a reduction in safety margin. No change in a safety limit or a safety limit margin is involved. We therefore conclude that the proposed changes to the Browns Ferry Technical Specifications with respect to the APRM and IRM systems are acceptable and do not involve a significant hazards consideration.

3.2 Snubbers

Table 3.6.H of the Browns Ferry Technical Specifications contains a list of "Shock Suppressors (snubbers)" that are required to be operable to protect the primary coolant system or other safety related components. Section 3.6.H.6 of the Technical Specifications states that: "Snubbers may be added to safety-related systems without prior license amendment to Table 3.6.H provided that a revision to Table 3.6.H is included with a subsequent license amendment request". TVA proposes to add three snubbers to Table 3.6.H on the Fire Protection System. They also propose to delete the two snubbers that were formerly on the control rod drive (CRD) line since the CRD return line has been capped at the reactor vessel and rerouted to the reactor water cleanup return line as part of the modifications to reduce the potential for cracking in the CRD return line. The line-and thus the snubbers-are no longer present in the system. TVA also proposes to delete four snubbers from Table 3.6.H on the condensate bypass line, since this line is a non-critical system (i.e., not classified as a safety-related system) and failure of this by-pass line will not cause damage to a critical system. We conclude that the proposed changes to Table 3.6.H are acceptable.

3.3 ASTM Procedure

Section 4.9.A.3 of the Technical Specifications requires that a sample of diesel fuel shall be analyzed once a month and that the quality shall be within the acceptable limits specified in an obsolete 1968 version of ASTM procedure D975. This ASTM procedure is under revision. In lieu of referring to the specific version of the ASTM procedure (which is subject to the periodic revisions) TVA has proposed to change the Technical Specifications to read: "The quality shall be within the acceptable limits specified in Table 1 of the latest revision to ASTM D975 and logged". Since the most recent revision to this standard method of analysis reflects the current best judgement of the country's experts who are on the various ASTM committees, the most recent edition of the standard is the one that should be used as the "referee method" rather than the edition in effect when the plant was under construction. We conclude that the proposed change to the Technical Specification is acceptable.

3.4 Rod Block Monitors

Control rod block functions are provided to prevent excessive control rod withdrawal so that the safety limit minimum critical power ratio is not violated. Two rod block monitor (RBM) channels are provided. The current Technical Specifications and the Bases therefore (Section 3.2.C.2) state that: "The minimum number of operable instrument channels specified in Table 3.2.C for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any thirty day period". TVA proposes to relocate this requirement in the Technical Specifications, adding it as part of "Note 7" to Table 3.2.C and rewording it to be more specific. The revised wording will be: "Two RBM channels are provided and only one of these may be out of service for testing and/or maintenance provided this condition does not last longer than 24 hours in any thirty day period". This is not a change to the requirements in the Technical Specifications but simply a change in wording of the requirement and its location in the Technical Specifications. We conclude that the proposed action is an improvement in phraseology and is acceptable.

3.5 Secondary Containment Testing

Section 4.7.C.b of the Technical Specifications required additional tests of secondary containment during the first operating cycle of each of the three Browns Ferry units to supplement the other specified tests which are conducted throughout the life of the plants. All three Browns Ferry units have completed their first operating cycle and the additional tests specified in Section 4.7.C.b. TVA, therefore, proposes to delete this requirement, since it is no longer applicable. We conclude that the proposed deletion is acceptable.

3.6 Milk Sample Locations

As part of the environmental radiological monitoring program at the Browns Ferry Nuclear Plant, TVA collects and analyzes a number of samples. The Browns Ferry Nuclear Plant Environmental Technical Specifications state that "milk shall be collected....from at least four farms in the vicinity of the plant..." and that"...any location from which milk can no longer be obtained may be dropped from the surveillance program. The NRC shall be notified in writing that milk-producing animals are no longer present at that location. An additional milk sampling location will then be added to the program..." (Section 4.2.3.b).

As of May 15, 1978, milk is no longer available from the dairy farm located approximately four miles north of Browns Ferry Nuclear Plant. The milk producing animals have been sold and removed from the farm. A dairy farm located approximately five miles north of the plant has been added to the monitoring program.

We have reviewed the meteorological data and deposition factors for the Browns Ferry plant and conclude that the new sample location is acceptable.

4.0 Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

5.0 Conclusion

We have concluded that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 16, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259, 50-260 AND 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 44 to Facility Operating License No. DPR-33, Amendment No. 40 to Facility Operating License No. DPR-52, and Amendment No. 17 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3, (the facility) located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

These amendments change the Technical Specifications to (1) permit the average power range monitor system to be inoperable in the refuel mode, provided the source range monitors are connected to give a non-coincidence, high flux scram; (2) permit less than three intermediate range monitors (IRMs) per trip channel to be operable in the shutdown or refuel modes, provided at least four IRMs (one in each core quadrant) are connected to give a non-coincidence, high flux scram; (3) clarifies ambiguous portions of the Technical Specifications related to the rod block monitor system; (4) removes reference to an obsolete 1968 version of an ASTM procedure; (5) modifies the list of snubbers that are required to be operable; (6) removes a specification for additional tests of secondary containment that only applied during the first fuel cycle for each Browns Ferry Unit, and (7) changes one of the four locations where milk samples are collected.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated August 2, 1978 and August 11, 1978, (2) Amendment No. 44 to License No. DPR-33, Amendment No. 40 to License No. DPR-52, and Amendment No. 17 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Athens Public Library, South and Forrest, Athens,

Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16 day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors