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Docket No. 50-260

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 35 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2. This Amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with cycle 2 operation of Unit No. 2. These changes are in response to your submittals dated October 28, 1977, March 10, 1978, March 22, 1978, April 20, 1978, May 26, 1978, June 1, 1978 and June 7, 1978. With your concurrence, we have modified the wording in Section 3.5.6.2 of your October 28, 1977 submittal to more precisely define the mode of operation permitted. Effective upon issuance of this Amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-52, is terminated.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 35 to DPR-52 – See Reports....
2. Safety Evaluation
3. Notice

MC020046199

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*SEE PREVIOUS YELLOW FOR CONCURRENCE

OFFICE	ORB#3	ORB#3	OELD	ORB#3		
SURNAME	*SSheppard	*RClark:acr	*ESilbersten	TIppolito		
DATE	6/15/78	6/15/78	6/19/78	6/ /78		

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Docket No. 50-260

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

Gentlemen:

The Commission has issued the enclosed Amendment No. to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2. This Amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with cycle 2 operation of Unit No. 2. These changes are in response to your submittal dated October 28, 1977, March 10, 1978, March 22, 1978, April 20, 1978, May 26, 1978, June 1, 1978 and June 7, 1978. With your concurrence, we have modified the wording in Section 3.5.6.2 of your October 28, 1977 submittal to more precisely define the mode of operation permitted. Effective upon issuance of this Amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-52, is terminated.

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-52
2. Safety Evaluation
3. Notice

EB Silbustein no legal objection with correction ~~CPR~~ to §3.3.4 of SER to indicate PCT to remain below ~2200°F. [illegible]

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME➤	SSheppard	RClark/acr	[illegible]	TIppolito		
DATE➤	6/15/78	6/15/78	6/19/78	6/21/78		

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

June 21, 1978

Docket No. 50-260

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 35 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2. This Amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with cycle 2 operation of Unit No. 2. These changes are in response to your submittals dated October 28, 1977, March 10, 1978, March 22, 1978, April 20, 1978, May 26, 1978, June 1, 1978 and June 7, 1978. With your concurrence, we have modified the wording in Section 3.5.6.2 of your October 28, 1977 submittal to more precisely define the mode of operation permitted. Effective upon issuance of this Amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-52, is terminated.

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

Sincerely,

TA Ippolito

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

Enclosures:
1. Amendment No. 35 to DPR-52
2. Safety Evaluation
3. Notice

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Ira L. Myers, M.D.
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U. S. Environmental Protection
Agency
Region IV Office
ATTN: EIS Coordinator
345 Courtland Street
Atlanta, Georgia 30308

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Tennessee Valley Authority (the licensee) dated October 28, 1977, March 10, 1978, March 22, 1978 and May 26, 1978, as supplemented by submittals dated April 20, 1978, June 1, 1978, and June 7, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 35, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 21, 1978

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-260

Introduction

By letter dated March 10, 1978, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-52 for the Browns Ferry Nucler Plant, Unit No. 2 (BFNP-2). Unit No. 2 shutdown on March 18, 1978 for the first refueling of the facility. During the outage, 132 of the 764 fuel assemblies in the core were replaced. Whereas the initial core consisted of all 7x7 fuel assemblies, the replacement fuel was 8x8 assemblies. The proposed changes to the Technical Specifications are to incorporate limiting conditions for operation associated with cycle 2 operation of Browns Ferry Unit No. 2, similar to those recently approved by the Commission for Browns Ferry Unit No. 1 (Amendment No. 35 to DPR-33 dated January 10, 1978), following replacement of 168 7x7 fuel assemblies in Unit No. 1 with 8x8 assemblies. Supplementary information was provided in TVA's letters dated June 1, 1978 and June 7, 1978.

By letter dated March 22, 1978, TVA submitted supplemental information to their March 10, 1978 submittal, a Loss-of-Coolant Accident (LOCA) analysis in response to the Commission's Order of March 11, 1977 and additional proposed changes to the Technical Specifications.

By letter dated May 26, 1978, TVA submitted additional proposed changes to the Technical Specifications to correct a discrepancy between the existing Technical Specifications and the analyses transmitted by their March 10, 1978 submittal. The analysis submitted on March 10 was based on having all eleven relief valves operable at all times, whereas it had previously been acceptable to have one of the 11 relief valves inoperable.

By letter dated October 28, 1977, TVA had requested changes to the Technical Specifications for Unit No. 1 to allow continuous operation with four of the six valves in the Automatic Depressurization System (ADS) operable; the current Technical Specifications require that five of the six valves be operable for continuous operation. In their letter of March 22, TVA requested that the same Technical Specifications proposed for Unit No. 1 be applicable to Unit No. 2. Additional information to support this request was supplied in a letter dated April 20, 1978.

During the current outage, TVA has modified Unit No. 2 to provide automatic trip of both recirculation pumps after turbine trip or generator load rejection. The purpose of this trip is to reduce the peak reactor pressure and peak heat flux resulting from transients and coincident failure of the turbine bypass system. The analysis of the recirculation pump trip (RPT) system was presented in TVA's March 10, 1978 submittal.

2.0 Discussion

In support of the reload application, the licensee provided the GE Reload 1 licensing submittal for BFNP-2 (Reference 1). However in contrast to the 168 fuel bundle reload size analyzed in the original submittal, TVA decided to use a 132 fuel bundle reload size. Subsequently General Electric performed a reanalysis based on the 132 reload size. The results of this reanalysis is reported in Reference 10. The reduction in the number of fresh fuel bundles was obtained by placing low burnup bundles in the outer cells, instead of fresh fuel bundles, and rearranging the inner core bundles such that the 132 bundles reload configuration would be less reactive than the 168 reload configuration. The operating limits proposed by TVA are conservatively based on the most limiting conditions determined by the two analyses (Reference 10). Technical Specification changes, information on the BPNP-2 Loss of Coolant Accident (LOCA) analysis, an increased relief valve simmer margin evaluation, and responses to NRC requests for additional information as listed in the reference section of this report were also provided by TVA.

The 132 bundles to be reloaded are GE 8x8 fuel. The description of the nuclear and mechanical design of the 8x8 fuel is contained in GE's licensing topical report for BWR reloads (Reference 6). Reference 6 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing 7x7, 8x8, and 8x8R fuel. Portions of the plant-specific data such as operating conditions and design parameters which are used in transient and accident calculations have also been included in Reference 6.

The staff's safety evaluation (Reference 8) of the GE generic reload licensing topical report has concluded that the nuclear and mechanical design of the 8x8 fuel, and GE's analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing 7x7, 8x8, and 8x8R fuel are acceptable. Approval of the nuclear and mechanical design of 8x8 fuel was originally based on information in Reference 7 and expressed in the staff's evaluation (Reference 9) of that document.

Based on the staff's review, the plant-specific input data for transient and accident analyses presented in Reference 6 are acceptable (Reference 8). Additional plant and cycle-dependent data and information are provided in References 1 and 10, which closely follow the outline of Appendix A of Reference 6.

Since the staff has reviewed and approved a large number of generic considerations related to use of reloads with mixed 8x8 and 7x7 fuel presented in Reference 8, only a limited number of additional areas of review have been included in this safety evaluation. The specific areas included are the plant and cycle-specific input data and results presented in References 1 and 10, the physics startup test program described in Reference 5, and those items identified in Reference 8 requiring special attention during reload reviews.

3.0 Evaluation

3.1 Nuclear Characteristics

For Cycle 2 operation of BFNP-2, 108 fresh 8x8 fuel bundles of type 8D274L and and 24 fresh 8x8 bundles of type 8D274H will be loaded into the core (Reference 10). The remainder of the 764 fuel bundles in the core will be 7x7 fuel exposed during the first cycle. The fuel loading pattern will be as shown in Figure 1A of Reference 10.

Based on the data presented in sections 4 and 5 of References 1 and 10, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 2.

3.2 Thermal Hydraulics

3.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 6, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients or from undetected fuel loading errors is 1.06. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as the result of the worst possible fuel loading error.

The most important of the plant/cycle-specific uncertainties, the TIP (transverse incore probe) uncertainty will be evaluated during the physics startup tests to confirm that the TIP uncertainty is within acceptable bounds (References 5 and 10).

2.2.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 2 operations. The most limiting operational transients and the fuel loading error have been analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR (References 1 and 10).

The transients evaluated were the generator load rejection without bypass, feedwater controller failure at maximum demand, loss of 100°F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6 and 7 and Figure 2 of References 1 and 10 were assumed.

The input data for the transient calculations have been reviewed and provide adequate conservatism for determination of transient reductions in CPR. The calculated system responses and reductions in CPR during each of the above transients are given in References 1 and 10. The largest reductions in CPR were for the generator load rejection without bypass and the control rod withdrawal error. For the load rejection, the ΔCPR values were 0.22 and 0.16, for the 8x8 fuel and the 7x7 fuel respectively. For the control rod withdrawal error, the ΔCPR values were 0.12 and 0.17, also respective to the 8x8 and 7x7 fuel.

Fuel loading errors have also been taken into account in Reference 1 and 10. Results of the analysis show that a mismatched 8x8 fuel bundle could provide a reduction in CPR of 0.25 for the 7x7 fuel, which exceeds the CPR associated with the above transients. This fuel loading error could, therefore, decrease the MCPR below the safety limit MCPR if the operating limit were based solely on the consideration of the transients.

Consideration of the most severe CPR errors or reductions to the safety limit (1.06) gives the appropriate operating limit MCPR. This results in a MCPR of 1.28 for 8x8 fuel and 1.31 for 7x7 fuel. These operating limit MCPRs will assure that the safety limit MCPR is not violated due to transients or fuel loading errors.

3.3 Accident Analysis

3.3.1 ECCS Appendix K Analysis

Input data and results for the BFNP-2 ECCS analysis are given in Reference 2. The information presented fulfills the requirements as outlined in Reference 8 and in 10 CFR 50.46 and Appendix K to 10 CFR 50.

We have reviewed the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Reference 2 and determined these values acceptable when BFNP-2 is operated at a MCPR equal to or greater than 1.20. The more restrictive MCPR limits based on the Fuel Loading Error (described above) is therefore more limiting than the Loss-of-Coolant-Accident event.

3.3.2 Control Rod Drop Accident

For the worst case control rod drop accident (CRDA) under hot startup conditions, the characteristic parameters for the accident meet the requirements for bounding analyses described in Reference 6. As stated in Reference 8, this is adequate to show that the design basis of 280 cal/gm peak fuel enthalpy for a hot startup CRDA is met. Because the characteristic accident parameters for the worst cold startup CRDA do not satisfy the requirements for bounding analyses, it was necessary to perform a plant specific analysis. The resulting calculated peak fuel enthalpy for the postulated CRDA was 179 cal/gm, which is acceptable.

3.3.3 Fuel Loading Error

As discussed in Section 2.3, potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been evaluated. The GE method for analysis has been reviewed and approved by the staff (Reference 9).

The analyses which have been performed for potential BFNP-2 fuel loading errors are acceptable for assuring that CPR's will not be below the safety limit MCPR of 1.06.

3.3.4 ADS Out-of-Service Analysis

The Automatic Depressurization System (ADS) is provided to aid in vessel depressurization following a small break loss-of-coolant accident (LOCA). Thus, the ADS only affects the results of break analyses where depressurization through the break itself is relatively slow (small breaks), and operation of the ADS increases the depressurization rate, allowing low pressure systems (such as the core spray (CS) and the low-pressure-coolant-injection (LPCI) systems) to reach higher flows sooner. This causes earlier reflood and lower calculated peak-cladding-temperature (PCT) results for the small break analyses. The more installed relief capacity (i.e., number of valves) in the ADS, the more pronounced is this effect.

Previous small break analyses, in the small-break-size range, where ADS has an appreciable effect (0 to approximately 0.5·ft^2) took credit for operation of five of the six ADS valves.(11) Maximum PCT in that break size range was around 1530°F, far below the larger (and limiting) break sizes whose PCT are around and still below 2200°F.

Continuous reactor operation with only four of the six ADS valves operable is acceptable if the small breaks' PCT does not exceed 2200°F for any fuel operating at the MAPLHGR limit.

The application for change in Technical Specifications (11) contained a generic estimate of a 200°F PCT increase for small breaks in the range affected by ADS capacity (0 to 0.5 ft^2). We required substantiation of that estimate for BFNP 1 and 2 in three areas, which was provided in reference 12 as discussed below.

1) The estimate of 200°F PCT increase was provided for the BRNP 1 and 2 plant type by a Generic ADS out-of-service analysis, which included calculations for a 251-inch inside-diameter pressure vessel, LPCI modified BWR/4(12), BFNP 1 and 2 are plants in this category), i.e., calculations were for the proper plant size and type.

2) The generic estimate of 200°F PCT increase was confirmed for the ADS steam flow range appropriate for BFNP 1 and 2 (with four and five ADS valves operable) by the Generic ADS out-of-service analysis, which included the BFNP 1 and 2 ADS' capacity range.

3) The model used for the Generic ADS out-of-service analysis did not contain the latest model changes described in reference 13. However, those model changes have not caused significant changes in the PCT results for the small break analyses of a smaller sized BWR/4 and an identical sized BWR/3,(2) and similarly the changes would not significantly affect small break PCT results for BFNP 1 and 2.

For other reasons, the model changes(13) llowed operation at slightly higher MAPLHGR limits. At these higher powers, small break PCT results could be as much as 140°F higher. Therefore, PCT for the worst small break with four of the six ADS valves operable would be approximately 1530°F + 200°F + 140°F = 1870°F. This is considerably below 2200°F and is therefore acceptable.

We, therefore, conclude that the material presented and discussed above adequately supports the TVA request to operate continuously with four of the six ADS valves in service, and such operation is, therefore, acceptable.

3.4 Overpressure Analysis

The BFNP-2 overpressure analysis for the MSIV (Main Steam Isolation Valve) closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 8. As specified in Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has been evaluated. Also, the method of applying the ATWS RPT features, corresponding to the current design, is to trip the recirculation system after the dome pressure has reached the specified setpoints. The effects of the ATWS RPT on transients generally result in an increase in the peak vessel pressure and a reduction in peak neutron average surface heat flux. As described in Reference 6, no credit is taken for the mitigating effects of ATWS RPT system in the establishment of thermal limits.

Based on the analysis and sensitivity studies submitted by the licensee and Reference 8, the overpressure analysis for BFNP-2 has been found acceptable.

3.5 Thermal-Hydraulic Stability

The results of the BFNP-2 thermal-hydraulic stability analysis (References 1 & 10) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the stability limit.

Also, the licensee has proposed restrictions on operating in the natural circulation mode. These restrictions, as proposed in the Technical Specifications, prohibit steady state operations without forced circulation for more than 12 hours, and the start of a recirculation pump from natural circulation unless the temperature difference between the loop started and the core coolant temperature is less than 75°F. We find the above analysis and proposed Technical Specification limits acceptable for Cycle 2 operations.

3.6 Recirculation Pump Trip

The licensee's analysis of the RPT system was evaluated on the basis of the evaluation in Reference 14. On the basis of our evaluation of the consequences of the failure of this trip input, we conclude that the RPT system is acceptable.

3.7 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 2. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload.

Methods and criteria for the tests have been described in References and are found acceptable. A written report of the startup tests will be provided to NRC in accordance with the requirements in Section 6.7 of the Technical Specifications.

4.0 Technical Specifications

The changes to the Technical Specification as proposed by TVA are acceptable with the following exceptions:

1) The operating limit MCPR for the 7x7 fuel shall be changed to 1.31. This change reflects an increase in MCPR such that the core wide safety limit will not be violated for the worst case fuel loading error.

2) The proposed wording in the Technical Specifications relating to the number of operable ADS valves shall more precisely define allowable operations with three or more ADS valves incapable of automatic operation.

5.0 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.

6.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

References

1. Letter, N. B. Hughes, (TVA) to E. G. Case (NRC) dated March 10, 1978 requesting Amendment to License and transmitting proposed Technical Specification changes and Supplemental Reload Analysis NEDO-24095.

2. Letter, L. M. Bills (TVA) to E. G. Case (NRC) dated March 22, 1978 transmitting proposed Technical Specification changes and LOCA Analysis NEDO-24088-1.

3. Safety Evaluation Report on Operation of Browns Ferry Units 1 and 2 With Four of The Six ADS Valves Operable, May 9, 1978.

4. Letter, R. H. Davidson (TVA) to E. G. Case (NRC) dated May 26, 1978.

5. Letter, J. E. Gilleland (TVA) to G. Lear (NRC) dated June 1, 1978.

6. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P, May 1977.

7. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.

8. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

9. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, April 1975.

10. Letter, J. E. Gilleland (TVA) to G. Lear (NRC) dated June 7, 1978.

11. Letter to E. Case, NRC, from J. Gilleland, TVA, October 28, 1977.

12. Letter to Director of NRR, Attention of G. Lear, From J. E. Gilleland, TVA, April 20, 1978.

13. Letter to G. Sherwood, General Electric Co., from K. Goller, NRC, SER for GE ECCS Evaluation Model, April 12, 1977.

14. Report to the Advisory Committee on Reactor Safeguards in the matter of Georgia Power Company, et al, Edwin I. Hatch Nuclear Plant, Unit No. 2, January 1978.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 35 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 2 (the facility) located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications to permit operation of the facility in the second fuel cycle, following the first refueling, during which 132 of the 764 fuel assemblies were replaced.

The applications for this amendment 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant

to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated October 28, 1977, March 10, 1978, March 22, 1978 and May 26, 1978, as supplemented by submittals dated April 20, 1978, June 1, 1978 and June 7, 1978, (2) Amendment No. 35 to License No. DPR-52, 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 21 day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

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