May 19, 1977

Docket No. 50-296

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Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 5 to Facility License No. DPR-68 for the Browns Ferry Nuclear Plant, Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your request of March 31, 1977, as supplemented April 21, 1977.

The amendment changes the Technical Specifications to allow replacement of either or both of the two Crosby reactor coolant system pressure relief valves with Target Rock valves of slightly smaller capacity provided that the Target Rock valves are set to relieve at a lower pressure.

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

Sincerely,

/s/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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

cc w/encl:
See next page

OFFICE	DOR:ORB#1	OELD	DOR:ORB#1		
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NRC FORM 318 (9-76) NRCM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1976 – 626-624

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

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Tennessee Valley Authority
ATTN: Mr. Godwin Williams, Jr.
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A. Schwencer, Chief
Operating Reactors Branch #1
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cc w/encl:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 5
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 31, 1977, as supplemented by submittal dated April 21, 1977, complies 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 application, 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-68 is hereby amended to read as follows:

"(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 5 , 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 19, 1977

ATTACHMENT TO LICENSE AMENDMENT NO.5

FACILIT LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

Remove pages 26, 27, 28, and 30 and replace with identically numbered pages.

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limts from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the pressure safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

Protective Action	Limiting Safety System Setting
A. Nuclear system safety valves open--nuclear system pressure	1,250 psig ± 13 psi (2 valves)
B. Nuclear system relief valves open--nuclear system pressure	
Target - Rocks	1,080 psig ± 11 psi (3 Or 4 valves) *
	1,090 psig ± 11 psi (3 or 4 valves) *

1.2 REACTOR COOLANT SYSTEM INTEGRITY

2.2 REACTOR COOLANT SYSTEM INTEGRITY

	1,100 psig ± 11 psi (3 valves)
Crosbys	1,110 psig ± 11 psi (2, 1, or 0 valves)*

* Total of 11 relief valves in combinations as indicated. See Bases.

C. Scram--nuclear system high pressure ≤ 1,055 psig

Amendment No. 5

1.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The safety limits for the reactor coolant system pressure have been selected such that they are below pressures at which it can be shown that the integrity of the system is not endangered. However, the pressure safety limits are set high enough such that no foreseeable circumstances can cause the system pressure to rise over these limits. The pressure safety limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The design pressure (1,250 psig) of the reactor vessel is established such that, when the 10 percent allowance (125 psi) allowed by the ASME Boiler and Pressure Vessel Code Section III for pressure transients is added to the design pressure, a transient pressure limit of 1,375 psig is established.

Correspondingly, the design pressure (1,148 psig for suction and 1,326 psig for discharge) of the reactor recirculation system piping are such that, when the 20 percent allowance (230 and 265 psi) allowed by USAS Piping Code, Section B31.1 for pressure transients are added to the design pressures, transient pressure limits of 1,378 and 1,591 psig are established. Thus, the pressure safety limit applicable to power operation is established at 1,375 psig (the lowest transient overpressure allowed by the pertinent codes), ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.

The Code Overpressure Protection Analysis (MSIV closure with Flux Scram) as described in a TVA letter to NRC of April 21, 1977, J. E. Gilleland to A. Schwencer, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1375 psig given in subsection 4.2 of the safety analysis report is well above the peak pressure event described above. Thus, the pressure safety limit applicable to power operation is well above the peak pressure that can result due to reasonably expected overpressure transients. The above TVA letter to NRC of April 21, 1977, demonstrates that the substitution of either or both of the Crosby valves by a Target Rock valve results in acceptable peak vessel pressures.

Higher design pressures have been established for piping within the reactor coolant system than for the reactor vessel. These increased design pressures create a consistent design which assures that, if the pressure within the reactor vessel does not exceed 1,375 psig, the pressures within the piping cannot exceed their respective transient pressure limits due to static and pump heads.

2.2 BASES

REACTOR COOLANT SYSTEM INTEGRITY

The pressure relief system for each unit at the Browns Ferry Nuclear Plant has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME Code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4-1 of subsection 4.4 which states that the nuclear system relief valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

Thirteen safety/relief valves have been installed on each unit with a total capacity of 78.7% of rated steam flow. The analysis of the worst overpressure transient, (closure of all main steam line isolation valves using 11 Target Rock Safety Relief Valves and 2 Dresser Safety Valves the most limiting configuration) neglecting the direct scram (isolation valve position scram), results in a peak nuclear system pressure at the bottom of the vessel of 1304 psig if a high neutron flux scram is assumed. The resulting 71 psig margin to the ASME Code limit of 1375 psig assures adequate protection against overpressurization. (Referenced from Sections 3.1.3 and 3.2.3 of NEDO-21165 of January 1976, entitled "GETAB Analysis Including the Effects of Neutron-Effective Voids and Substitution of Crosby Valves - BFNP Unit 3," modified by TVA letter to NRC dated April 21, 1977, J. E. Gilleland to A. Schwencer).

To meet the second design basis, the total safety/reliefcapacity of 78.7% has been divided into 64.5% relief (11 valves) and 14.2% safety (2 valves). The analysis of the most severe abnormal operational transient resulting in a nuclear system pressure increase (turbine trip from high power without bypass) assuming that the scram is initiated by the position switches on the turbine stop valves is presented in Section 3.2.1 of NEDO-21165 of January 1966, entitled, "GETAB Analysis Including the Effects of Neutron-Effective Voids and Substitution of Crosby Valves - BFNP Unit 3," modified by TVA letter to NRC dated April 21, 1977, J. E. Gilleland to A. Schwencer.) This analysis shows that the relief valves limit pressure at the safety valves to 1178 psig which is 72 psig below the spring safety valve setpoint. This analysis also shows that the peak nuclear system pressure is 1219 psig at the bottom of the vessel which is 156 psig below the allowed ASME Code limit of 1375 psig.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 5 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

DOCKET NO. 50-296

Introduction

Tennessee Valley Authority (TVA) submitted an application for amendment to Operating License No. DPR-68 on March 31, 1977, which proposed changes to the Browns Ferry Nuclear Plant, Unit No. 3 Technical Specifications. The proposed changes provide for the replacement of one or both Crosby relief valves with Target Rock relief valves set to relieve at a lower pressure.

Background

The overpressure protection features for the reactor vessel and other portions of the reactor coolant system (RCS) were described in Section 4.9 of the Safety Evaluation Report of the TVA Browns Ferry Nuclear Plant Units 1, 2, and 3, dated June 26, 1972, and were found to be acceptable. The design consisted of two safety valves (Dresser safety valves) and eleven relief valves (Target Rock relief valves) similar in design to valves used on other General Electric (GE) boiling water reactor plants . The acceptance was based upon an overpressure protection report prepared in accordance with the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The safety valves and the relief valves when acting as pressure relief devices were sized to limit the RCS pressure to less than the pressure allowed by the Code when the main steam isolation valves are tripped closed. This transient is the most severe RCS overpressure transient for the condition of this analysis.

For Browns Ferry Unit No. 3, TVA elected to replace two of the eleven Target Rock relief valves with Crosby relief valves (which were of a new design) and to change the setpoint pressures for both the safety and relief valves. The GE report NEDO-21165, "GETAB Analysis Including the Effects of Neutron-Effective Voids and Substitution of Crosby Valves - Browns Ferry Nuclear Plant Unit 3," dated January 1976, presented the results of a revised overpressure protection analysis. As before, the main steam isolation valves closure transient was the most severe RCS overpressure transient for the conditions of this Code analysis. The analysis found the peak RCS pressure to be within that allowed by the Code for the transient, with a margin of 79 psi between calculated and Code allowable pressure for the reactor vessel.

In NEDO-21165, the General Electric Company described the sensitivity of the peak RCS pressure to failure of a single valve and concluded that this failure would result in less than a 20 psi increase, a pressure increment well within the margin indicated above. This description of the impact of a failed valve was based upon a generic sensitivity study previously reported to the Commission in GE report "Code Overpressure Protection-Sensitivity of Peak Vessel Pressure to Valve Operability," dated January 23, 1975.

In the course of our review, we raised questions concerning the application of this generic sensitivity study to the Browns Ferry Unit No. 3 overpressure protection analysis. TVA submitted satisfactory responses to our requests on this matter in a letter dated April 21, 1977.

Evaluation

In TVA-BFNP-TS-81 dated March 31, 1977, TVA described the difference in the relieving capacity and actuation time between Crosby and Target Rock valve design as negligible and concluded the analysis of the overpressure protection margin presented in NEDO-21165 for the valve configuration of 9 Target Rock valves and 2 Crosby valves bounds the other possible cases in which one or both of the Crosby valves are replaced with Target Rock valves at lower setpoints. Since it was not immediately clear from the described TVA bases that the TVA conclusion was valid, we requested TVA to provide a sensitivity study of the peak pressure for 1) relief valve delay time 2) relief valve set point, and 3) relief valve capacity to verify TVA's conclusion. TVA responded to this request in a letter dated April 21,1977.

Our review of the information in this letter found that the valve configuration of 9 Target Rocks and 2 Crosby does not bound the other possible configurations proposed by TVA. However, we have concluded that the difference between these configurations is not significant and that analysis shows that the peak RCS pressure will remain within that allowed by the Code for the transient. The margin between calculated and Code allowable pressures for the reactor vessel is 71 psi for the most limiting valve configurations. The results of the analysis conform with the criteria used for acceptance of the design for overpressure protection features. We conclude, therefore, that the modifications to the overpressure protection features, and the safety and relief valve setpoint pressures proposed by TVA are acceptable and can be incorporated into the Technical Specifications.

Environmental Consideration

We have determined that the amendment does 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 the amendment involves 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 this amendment.

Conclusion

We have concluded, based on the consideration 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.

Date: May 19, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 5 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 No. 3 (the facility) located in Limestone County, Alabama. The amendment is effective of as the date of issuance.

The amendment changes the Technical Specifications to allow replacement of either or both of the two Crosby reactor coolant system pressure relief valves with Target Rock valves of slightly smaller capacity provided that the Target Rock valves are set to relieve at a lower pressure.

The application for the amendment complies 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 application for amendment dated March 31, 1977, as supplemented April 21, 1977, (2) Amendment No. 5 to License No. DRP-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, NW., 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 19th day of May 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors