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SUBJECT: Forwards pages inadvertently left out of Amend 6 to updated FSAR for Browns Ferry Nuclear Plant.
Corrected 11/14/88

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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 - CORRECTIONS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) - AMENDMENT NO. 6.

Reference: TVA letter to NRC dated July 19, 1988, "Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 - Updated Final Safety Analysis Report - Amendment No. 6"

In the reproduction of the BFN-UFSAR Amendment No. 6 the enclosed pages were inadvertently left out. An instruction sheet has also been included for incorporating this material into your copies of the BFN-UFSAR.

If you have any questions, please telephone Raymond L. Boyd at (205) 729-2854.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. Gridley, Manager
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Regulatory Affairs

Enclosure
cc: See page 2

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U.S. Nuclear Regulatory Commission

SEP 19 1988

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INSTRUCTION SHEET

Remove	Insert
3.4-13/3.6-14 3.6-13/14	3.4-13/14 3.6-13/14
--/--	9.0-i/--

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

Pumps

One 100% capacity supply pump is provided for each unit to pressurize the system with water from the condensate storage tank. One 100% capacity spare pump is provided for Units 1 and 2. It can supply water to either or both control rod drive hydraulic systems. Unit 3's system is separate and has one spare 100% capacity pump. Change over of the pumps is manual locally or from the main control room. Each pump is installed with a suction strainer and a discharge check valve to prevent bypassing flow backwards through the nonoperating pump.

A minimum flow bypass connection between the discharge of the pumps and the condensate storage tank prevents overheating of the pumps in the event that the pump discharge is inadvertently closed. In addition, a portion of the CRD flow is directed to the recirculation pump seals for pump seal cooling. Pump discharge pressure is indicated at the pump by a pressure indicator.

Filters

Two parallel filters remove foreign material larger than 50 microns absolute (25 microns nominal) from the hydraulic supply subsystem water. The isolated filter can be drained, cleaned, and vented for reuse while the other is in service. A differential pressure indicator and alarm monitor the filter element as it collects foreign material. A strainer in the filter discharge line guards the hydraulic system in the event of filter element failure.

Accumulator Charging Pressure

The accumulator charging pressure is maintained automatically by a flow-sensing element, controller, and an air-operated flow control valve. During normal operation, the accumulator charging pressure is established upstream from the flow control valve by the restriction of the flow control valve. During scram, the flow-sensing system upstream of the accumulator charging header detects high flow in the charging header and partly closes the flow control valve. The flow control valve is closed enough so that the proper flow to recharge the accumulators is diverted from the hydraulic supply header to the accumulator charging header. The parallel spare valve is provided with isolation valves to permit maintenance of the noncontrolling valve.

The pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm. | 6

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The pressure in the charging header is monitored in the control room with a pressure indicator and low pressure alarm.

During normal operation, the constant flow established through the flow control valve is the sum of the maximum water required to cool all the drives and that amount of water needed to provide a stable hydraulic system for insertion and withdrawal of the mechanism.

Drive Water Pressure

The drive water pressure control valve, which is manually adjusted from the control room, maintains the required pressure in the drive water header.

A flow rate of approximately 6 gpm (the sum of the flow rates required to insert and to withdraw a control rod) normally passes from the drive water pressure header through two solenoid-operated stabilizing valves (arranged in parallel) and then goes into the line downstream from the cooling pressure control valve. One stabilizing valve passes flow equal to the drive insert flow; the other passes flow equal to the drive withdrawal flow. The appropriate stabilizing valve is closed when operating a drive to divert the required flow to the drive. Thus, the flow through the drive pressure control valve is always constant.

Flow indicators are provided in the drive water header and in the line downstream from the stabilizing valves, so that flow rate through the stabilizing valves can be adjusted. Differential pressure between the reactor vessel and the drive water pressure header is indicated in the control room.

Cooling Water Pressure

The cooling water header passes the flow from the drive water pressure control valve through the control rod drives and into the vessel. At normal flow rates, the cooling water header pressure will be approximately 20 psi above reactor vessel pressure.

A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and the drive cooling water pressure. Although the drives can function without cooling, the life of their seals is shortened by exposure to reactor temperatures.

Exhaust Water Header

The exhaust water header takes water discharged during a normal control rod positioning operation and directs it through the other CRD exhaust valves into the reactor vessel. If necessary, the exhaust water may be directed into the reactor vessel via the RWCU system by opening a normally closed valve.

oscillations is of the order of hours, the effect of neglecting delayed neutrons is small.

c. Linearized first order perturbation theory is adequate. Any use of linearized equations coupled with first order perturbations implies small perturbations for consistency, and determines the response of the system to small changes. Near the oscillation threshold the assumption of small perturbations is valid to determine the response of the reactor.

d. Mode coupling is negligible. Mode coupling arises from the expansion of the perturbation amplitudes in orthogonal eigen-functions of the unperturbed boundary value problem and the multiplication by the orthogonal eigen-functions followed by integration over the volume. The cross mode terms are of the form and are small because of the orthogonality of the g 's. They are zero if the flux is flat; furthermore, Canosa has shown that their neglect has little effect in the flux ranges of interest for boiling water reactor design.

e. Neturon absorption in iodine is neglected (<< λ). The assumption that the absorptions in iodine are negligible corresponds closely to the physical situation; however, it affects the normalization factor to be used in the xenon and iodine governing differential equations. The estimated accuracy in determining threshold fluxes, oscillation periods, and damping ratios is of the order of 10%.

The important parameters used to predict stability against xenon-induced power oscillations are the core length-to-diameter (L/D) ratio, flux or power level, power coefficient, percent of power flattening, and migration area or void transportation in axial direction. The relative stability of any given reactor with respect to power disturbance effects depends on the length-to-diameter (L/D) ratio of the core. Shown in Figure 3.6-9 is the relative stability of the radial, azimuthal, and axial power distributions as a function of L/D ratio. Figure 3.6-9 shows that axial and azimuthal oscillations have a much higher probability of occurrence than radial oscillations for the reactor. This plot only gives information regarding the relative stability of the various power distributions for a given core. The effect of the flux or power level on the stability of the axial power distribution is shown in Figure 3.6-10. These data, which were adopted from Lellouche, include the effect of void transport in the axial direction. Figure 3.6-10 shows that the reactor is well damped, and that there is a factor of 4 between the threshold power coefficient in which xenon oscillations would be induced and the calculated power coefficient at the end of the reactor cycle.

Flux flattening tends to reduce the stability of the various geometrical modes. This is due to the fact that the buckling differences, and consequently the criticality differences between the harmonics, are reduced as the flux shape is flattened. This flux flattening may cause the radial power distribution to be subject to azimuthal oscillations (a rotating edge-to-edge tilt) which are less well damped than the axial oscillations. Shown in Figure 3.6-11, which is based on work by Randall and St. John are the critical power coefficients, where the onset of neutral oscillation occurs, as a function of flux flattening. From this plot it is seen that the reactor is well damped even when the flat zone is 100 percent of the radius and demonstrates that the stability requirements for the plant are satisfied.

3.6.6 Nuclear Evaluations

The analyses presented in paragraph 3.6.5 show that the safety design basis is satisfied in conjunction with the nuclear design requirements of paragraph 3.6.4. Adequate protection is provided for the cladding and the nuclear system process barrier. The nuclear requirements for reactivity control systems and the settings of the reactor protection system are primarily associated with limitations on the levels and rates of change of reactivity, power, and temperature. Normal plant operation is conducted at rates and values of these parameters such that reactor transients are readily observable and controllable by plant personnel.

The reactor protection system responds to some abnormal operational transients by initiating a scram. The reactor protection system and the control rod drive system act quickly enough to prevent the initiating distrubance from causing fuel damage. The scram reactivity curve used in the reactivity excursion analyses for the initial core is shown in Figure 3.6-12. The scram reactivity for termination of these abnormal operational transients is shown in Figure 3.6-13. Figures showing the scram reactivity curves for the current operating cycle are given in the applicable reload amendment. Abnormal operational transients are evaluated in Section 12, 'Plant Safety Analysis.' No fuel damage results from any abnormal operational transient.

The specified rod withdrawal sequences and the rod worth minimizer maintain rod worth at acceptable low values to minimize the consequence of a reactivity accident. At any specified reactor state, peak enthalpies for rod removal accidents vary directly with rod worths; it is the peak enthalpy which is the best measure of the consequences of a reactivity accident. The uranium dioxide vapor pressure data of Ackerman and the interpretation of all the available tests in the TREAT facility of Argonne National Laboratory indicate that the sudden fuel pin rupture threshold is about 425 calories per gram. Analyses indicate that prompt dispersal of finely fragmented fuel into the geometrical modes. This is due to the fact that the buckling differences, and consequently the criticality differences between the harmonics, are reduced as the flux shape is flattened. This flux flattening may cause the radial power distribution to be subject to azimuthal oscillations (a rotating edge-to-edge tilt) which are less well damped than the axial oscillations. Shown in Figure 3.6-11, which is based on work by Randall and St. John are the critical power coefficients, where the onset of neutral oscillation occurs, as a function of flux flattening. From this plot it is seen that the reactor is well damped even when the flat zone is 100 percent of the radius and demonstrates that the stability requirements for the plant are satisfied.

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