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NOTE: Correct all copies of the applicable publication as specified below.

ITEM	REFERENCES (SECTION, PAGE PARAGRAPH, LINE)	INSTRUCTIONS (CORRECTIONS AND ADDITIONS)
01	Page v/vi	Replace with new page v/vi
02	Page 3-1/3-2	Replace with new page 3-1/3-2
03	Page 4-3	Replace with new page 4-3
04	Page 4-10	Replace with new page 4-10
05	Page 6-2	Replace with new page 6-2

LIST OF TABLES

3. INPUT INTO ANALYSIS

A list of the significant plant input parameters to the LOCA analysis is presented in Table 1.

Table 1
SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core Thermal Power	3435 MWt, which corresponds to 105% of rated steam flow
Vessel Steam Output	14.09×10^6 lbm/h, which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks - Discharge	1.9 ft^2 (DBA) 1.3 ft^2 (66% DBA)
- Suction	4.1 ft^2
Number of Drilled Bundles	764

Fuel Parameters:

	Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio*
A.	IC Type 1	8x8	13.4	1.4	1.2
B.	IC Type 2	8x8	13.4	1.4	1.2
C.	8DRB265L	8x8	13.4	1.4	1.2
D.	P8DRB299	8x8	13.4	1.4	1.2
E.	P8DRB284Z	8x8	13.4	1.4	1.2
F.	P8DRB283 (LTA)	8x8	13.4	1.4	1.2
G.	P8DRB314 (LTA)	8x8	13.4	1.4	1.2
H.	BP8DRB284L	8x8	13.4	1.4	1.2

*To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

4.5 RESULTS OF THE CHASTE ANALYSIS

This code is used, with suitable inputs from the other codes, to calculate the fuel cladding heatup rate, peak cladding temperature, peak local cladding oxidation, and core-wide metal-water reaction for large breaks. The detailed fuel model in CHASTE considers transient gap conductance, clad swelling and rupture, and metal-water reaction. The empirical core spray heat transfer and channel wetting correlations are built into CHASTE, which solves the transient heat transfer equations for the entire LOCA transient at a single axial plane in a single fuel assembly. Iterative applications of CHASTE determine the maximum permissible planar power where required to satisfy the requirements of 10CFR50.46 acceptance criteria.

The CHASTE results presented are:

- Peak Cladding Temperature versus time
- Peak Cladding Temperature versus Break Area
- Peak Cladding Temperature and Peak Local Oxidation versus Planar Average Exposure for the most limiting break size
- Maximum Average Planar Heat Generation Rate (MAPLHGR) versus Planar Average Exposure for the most limiting break size.

A summary of the analytical results is given in Table 2. Table 3 lists the figures provided for this analysis. The MAPLHGR values for each fuel type in the BF-3 core are presented in Tables 4A through 4H.

4.6 METHODS

In the following sections, it will be useful to refer to the methods used to analyze DBA, large breaks, and small breaks. For jet-pump reactors, these are defined as follows:

a. <u>DBA Methods.</u> LAMB/SCAT/SAFE/DBA-REFLOOD/CHASTE. Break size: DBA.

b. Large Break Methods (LBM). LAMB/SCAT/SAFE/non-DBA REFLOOD CHASTE. Break sizes: 1.0 $ft^2 \leq A < DBA$.

c. Small Break Methods (SBM). SAFE/non-DBA REFLOOD. Heat transfer coefficients: nucleate boiling prior to core uncovery, 25 $Btu/hr\text{-}ft^2\text{-}{}^{o}F$ after recovery, core spray when appropriate. Peak cladding temperature and peak local oxidation are calculated in non-DBA-REFLOOD. Break sizes $A \leq 1.0\ ft^2$.

Table 4E

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: P8DRB284Z

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1695	0.004
1,000	11.2	1695	0.004
5,000	11.7	1744	0.005
10,000	12.0	1756	0.005
15,000	12.0	1762	0.005
20,000	11.9	1762	0.005
25,000	11.3	1727	0.004
30,000	10.8	1692	0.004
35,000	10.4	1646	0.003
40,000	9.9	1600	0.003
45,000	9.5	1546	0.002

Table 4F

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: P8DRB283 (LTA)

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1666	0.004
1,000	11.2	1669	0.004
5,000	11.7	1710	0.004
10,000	12.0	1715	0.004
15,000	12.0	1714	0.004
20,000	11.9	1706	0.004
25,000	11.3	1677	0.004
30,000	10.8	1646	0.003
35,000	10.4	1610	0.003
40,000	10.0	1568	0.002
45,000	9.5	1527	0.002

Table 4G

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: P8DRB314 (LTA)

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	10.6	1615	0.003
1,000	10.7	1621	0.003
5,000	11.3	1660	0.003
10,000	11.7	1693	0.004
15,000	11.5	1700	0.004
20,000	11.2	1698	0.004
25,000	10.6	1672	0.004
30,000	10.1	1626	0.003
35,000	9.7	1584	0.002
40,000	9.3	1547	0.002
45,000	8.8	1501	0.002

Table 4H

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: BF-3 Fuel Type: BP8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200	11.2	1694	0.004
1,000	11.3	1695	0.004
5,000	11.8	1744	0.005
10,000	12.0	1751	0.005
15,000	12.0	1758	0.005
20,000	11.9	1751	0.005
25,000	11.3	1699	0.004
30,000	10.8	1645	0.003
35,000	10.1	1576	0.002
40,000	9.4	1507	0.002
45,000	8.8	1439	0.001

6. CONCLUSIONS

Analyses have demonstrated that failure of the LPCI is the most severe failure among the low pressure ECCS because, unlike the core spray which must pass through the CCFL regions at the top of the core, LPCI is injected into the lower plenum through the jet pumps. Thus, the LPCI injection valve is the worst single failure in the large break region. This is the case for a break occurring in the either the suction or discharge piping. For a break in the discharge piping, this failure results in no LPCI flow, and for suction line failure, LPCI flow is minimized.

Comparison of the calculated PCT's for the maximum size break in the suction and discharge piping determines which is the DBA and which is the second most limiting location. For BF-3, the discharge break is the most limiting location (Table 2). The characteristics that determine which is the most limiting break area at the DBA location are:

a. the calculated hot node reflooding time

b. the calculated hot node uncovery time, and

c. the time of calculated boiling transition.

The time of calculated boiling transition increases with decreasing break size, since jet pump suction uncovery (which leads to boiling transition) is determined primary by the break size for a particular plant. The calculated hot node uncovery time also generally increases with decreasing break size, as it is primarily determined by the inventory loss during the blowdown. The hot node reflooding time is determined by a number of interacting phenomena such as depressurization rate, counter current flow limiting and a combination of available ECCS.

The period between hot node uncovery and reflooding is the period when the hot node has the lowest heat transfer. The break that results in the longest period during which hot node remains uncovered usually results in the highest calculated PCT. If two breaks have similar times during which the hot node remains uncovered, then the larger of the two breaks will be limiting as it

would have an earlier boiling transition time (i.e., the larger break would have a more severe LAMB/SCAT blowdown heat transfer analysis).

Figures 6a and 6b show the variation with discharge and suction break size of the calculated time the hot node remains uncovered for Browns Ferry Unit 3. The CHASTE calculation for the discharge DBA used the discharge DBA LAMB/SCAT and SAFE/REFLOOD results. The CHASTE calculation for the suction DBA used the suction DBA LAMB/SCAT and SAFE/REFLOOD results. In accordance with the conservative approach used for the lead plant, the 80% discharge DBA LAMB/SCAT results were used with the 66% discharge DBA SAFE/REFLOOD results to determine the 66% discharge DBA results.

Based on these results, the 0.66 DBA was determined to be the break that results in the highest calculated PCT in the 1.0 ft^2 to DBA region. The determination of the 0.66 DBA being the most limiting break was based on the reasoning discussed above. The detailed calculations for the lead plant (Reference 11) confirmed that this procedure for determining the most limiting break is justified. These results are shown in Figures 1c through 5c and were used to determine the MAPLHGR's in Tables 4A through 4H.

The discharge DBA results are shown in Figures 1b through 5b.

The second most limiting location for the LOCA is the recirculation suction line. The results of the maximum break in this piping (suction DBA) are shown on Figures 1a through 5a.

In Reference 14, the small break spectrum calculations were re-evaluated to include a DC power source failure. Figure 2 of Reference 14 is applicable to the small break spectrum for BF-3 with LPCI modification assuming one ADS valve out of service (BF-3 has 6 ADS valves). The limiting small break yields a PCT of 1769°F.

Table 6 of Reference 14 demonstrates the effect of the loss of an additional ADS valve (total of 2 ADS valves lost). The resultant PCT is given as 1924°F. This PCT is higher than the limiting large break PCT discussed earlier in this report (see Table 2, page 4-5) due to the extra conservatism of the two ADS valves out

DETERMINATION OF NO SIGNIFICANT HAZARDS

BROWNS FERRY NUCLEAR PLANT

<u>Description of amendment request</u>:

The amendment would revise the Technical Specifications (T.S.) of the operating license to: (1) modify the core physics, thermal and hydraulic limits to be consistent with the reanalyses associated with replacing about 1/3 of the core during the current refueling outage for unit 3 and (2) reflect plant modifications performed during the current refueling and modification outage. Specifically, the amendment would result in changes to the T.S. in the following thirteen areas:

1. Changes to the license related to the Cycle 6 core reload involving removal of depleted fuel assemblies in about one-third of the nuclear reactor core and replacement with new fuel of the same type previously loaded in the core with attendant license changes in the core protection safety limits and reactor protection system setpoints. The actual changes are slight adjustment (by 0.01 in initial core life) in the Operating Limit Minimum Critical Power Ratio (OLMCPR), an added table on maximum average planar linear heat generation rate (MAPLHGR) versus average planar exposure, a change to the upscale flow bias rod block, and RBM upscale for core flow to 105 percent and a change to the references in the bases to reflect that TVA performed the reload transient analysis.

2. Changes in the T.S. to reflect modifications to the torus as part of the Mark I containment program. They include;

   a. revising the tables listing surveillance instrumentation for suppression pool bulk temperature reflecting the installation of 16 sensors for an improved torus temperature monitoring system and a revision to the basis for the existing limits on torus water temperature;

   b. reducing the minimum required flow for two LPCI pumps in the same loop from 15000 gpm to 12000 gpm as a result of the installation of an orifice plate to eliminate vibration in the RHR return line;

   c. eliminating the requirement of placing the reactor in cold shutdown and performing a torus inspection after extended operation of relief valves above a suppression pool temperature of 130°F;

   d. revise the bases to reflect changes in the torus configuration.

3. Changes to the T.S. to reflect modifications to the scram discharge volumes (SDV), and the addition of a second scram discharge instrument volume (SDIV); each of the SDIVs now have redundant vent and drain valves and new, diverse level instrumentation. The changes to the T.S. are to add operability, surveillance and calibration requirements on the new level instrumentation and valves.

4. Changes to T.S. surveillance instrumentation tables to add new instrumentation for containment high-range radiation monitors and to add new instrumentation; and delete current instrumentation for drywell pressure-wide range and suppression chamber wide-range water level in response to requirements in NUREG-0737; items II.F.1.3, II.F.1.4 and II.F.1.5.

5. Changes to T.S. RPS instrumentation requirement tables to delete the bypass function if reactor pressure is less than 1055 psig and the mode switch not in the RUN mode.

6. Revisions to the T.S. table for containment isolation valve surveillance to reflect the revised location of several H_2/O_2 isolation valves.

7. Changes to T.S. surveillance instrumentation tables to reflect new instrument numbers for the new upgraded drywell temperature and pressure instrumentation.

8. Revisions to the table of testable penetrations to reflect the new testable penetrations as a result of modifications to make the flange side of several isolation valves testable.

9. Revision of the T.S. table for containment isolation valve surveillance to add two new isolation valves that are part of a newly installed redundant discharge line from the drywell compressor into containment and to delete one isolation valve which was removed from the demineralized water system.

10. Revision of T.S. to provide limiting conditions for operation and surveillance requirements for electric power monitoring for the reactor protection system power supply.

11. Modify the T.S. to apply to the new analog (continuous measuring) instrumentation. The analog instrumentation replaces certain mechanical-type pressure and level switches with a more accurate and more stable electronic transmitter/electronic switch system and will provide improved performance of trip functions for reactor protection system actuation, and containment isolation. The changes to the T.S. include:

    a. in the tables on functional test frequencies, calibration frequencies and surveillance requirements, for each switch replaced, add the instrument number and type of sensor beneath the parameter being monitored and/or controlled.

    b. add notes to the above tables to specify how the functional and calibration tests are to be conducted.

    c. in addition to the above administrative changes, the calibration requirements have been changed to incorporate extended calibration intervals. However, the required setpoints, functional test frequencies and channel check frequencies for the instrumentation will not be changed. The new calibration requirements, together with the new instrumentation, are expected to provide a more reliable instrumentation system.

12. Change the T.S. to reflect the addition of a thermal power monitor. The purpose of this monitor is to have the Average Power Range Monitor (APRM) flow biased neutron flux signal respond to the thermal flux rather than the neutron flux in the core by accounting for the approximately 6-second thermal time constant of the fuel. The specific changes to the T.S. are:

    a. Add the words "flow biased" in parenthesis to the heading for the limits on "APRM Flux Scram Trip Settings" to indicate that the settings are reduced according to the equations given in this section when there is less than 100% core flow.

    b. There is a trip unit separate from the APRM flow-biased scram at less than 120% instantaneous neutron flux. A new requirement is being added to require that whenever the mode switch is in the run position, the APRM fixed high flux scram trip setting shall be operable and set at S $\leq$ 120% power.

    c. The bases for the neutron flux scram are revised to describe the functions of the APRM Flow-Biased High Flux Scram Trip Setting and the Fixed High Neutron Flux Scram Trip.

    d. Since there is now a new trip system, the tables listing the operability requirements and functional test frequencies on the scram instrumentation have to be revised to add this new instrumentation.

13. Administrative changes to the T.S. involving changes to the Table of Contents to reflect the above license changes and miscellaneous editorial changes.

Bases for proposed no significant hazards consideration determination:

The Commission has provided guidance concerning the application of the standards by providing examples of actions that are likely, and are not likely, to involve significant hazard considerations (48 FR 14870). Four examples of actions not likely to involve significant hazards considerations are:

"(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.

(iii) For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable...

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

Each of the thirteen changes to the T.S. described previously is encompassed by one of the above examples of actions not likely to involve a significant hazards consideration. The basis for this determination on each of the thirteen changes is discussed below.

1. <u>Core Reload</u>

1.a Fuel Changes

The changes to the T.S. associated with removing depleted spent fuel from the reactor and replacing these with new fuel assemblies is encompassed by example (iii) above of those actions not likely to involve a significant hazards consideration.

The proposed reload involves fuel assemblies of the same type as previously found acceptable by the staff and loaded in the core in previous cycles. The analytical methods used by the licensee to demonstrate conformance to the technical specifications have been previously approved by the staff. In addition, no changes have been made to the acceptance criteria for the technical specification changes involved.

Since the replacement fuel assemblies are of the same type previously added to all three Browns Ferry units and other BWRs and since the codes, models and analytical techniques used to analyze the reload have been generically approved by the NRC, the changes to the T.S. associated with the reload are clearly encompassed by example (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

1.b Increased Core Flow

The licensee has evaluated normal and anticipated operational transients and accidents (e.g., rod drop accident) using methods previously reviewed and approved by the staff. The licensee has concluded that the proposed change will not result in an increase in the probability or consequences of previously analyzed accidents. The licensee has also analyzed the most limiting events to determine which event could potentially induce the largest reduction in the initial critical power ratio.

For BWRs such as Browns Ferry, the staff has established a safety limit minimum critical power ratio (MCPR) of 1.07. Operation above this limit precludes significant fuel failure. To insure that during operation the MCPR will not drop below the safety limit, licensees are required to analyze those transients and accidents which are most likely to significantly affect the critical power ratio coincident with possible failure of certain equipment to function as designed. The licensee is required to add the maximum calculated change in critical power ratio to the safety limit MCPR to establish an operating limit MCPR. Operating at or above this limit insures that even during postulated transients, the MCPR will not drop below the safety limit established by the staff. The licensee has performed the required analyses and proposed new operating limit MCPRs accordingly. Specifically, the licensee has proposed a slight increase in the operating limit minimum critical power ratio and proposed clipping the rod block monitor upscale flow

biased setpoint of 106 percent rated power to ensure adequate protection in the event of a rod withdrawal error. This increase in MCPR and clipping the rod block monitor upscale flow biased setpoint will increase the margin of safety. Thus the licensee concludes that there will be no reduction in the margin of safety established by the staff.

Based on the evaluation performed by the licensee and the fact that the proposed change is encompassed by example (vi) which is not likely to pose a significant hazards consideration, it is proposed to determine that the application for amendments involves no significant hazards considerations.

1.c References in the Bases

The changes in the T.S. associated with changing the references in the Bases to reflect that the reload transient analysis is now being performed by TVA is encompassed by examples (i) and (iii) above of those actions not likely to involve a significiant hazards consideration.

The reload analysis, in the past, has been performed by General Electric Company. This reload analysis has been performed by TVA using analytical methods described in TVA-TR81-01-A. The analytical methods have been approved by the staff. Since NRC has previously found these methods acceptable and the T.S. changes are being made

to achieve consistency between the methods used and the references in the Bases, these changes to the T.S. are clearly encompassed by examples (i) and (iii) of the guidance provided by the Commission for an action not likely to involve a significant hazards consideration.

2. Changes Related to Torus Modifications

The Commission issued an Order in the matter of Browns Ferry Unit 3 requiring completion-during the current refueling outage-of the plant modification required by the Mark I program so as to comply with the Staff's Acceptance Criteria contained in Appendix A to NUREG-0661. Numerous modifications are being implemented in the Unit 3 torus during the reload 5 refueling outage as part of the Mark I Containment Program. These modifications are required by NRC to restore the originally intended margins of safety in the containment design. Most of the major internal structural modifications to the torus were completed during the previous refueling outage. These modifications are discussed in Amendment No. 51 to Facility Operating License No. DPR-68 issued March 24, 1982. The modifications being made during this outage will complete the requirements specified in NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program."

One of the changes to the T.S. is to revise the tables that list the surveillance instrumentation associated with the suppression pool bulk temperature. This modification provides an improved torus temperature monitoring system which consists of 16 sensors. This will provide a more

accurate indication of the torus water bulk temperature as required by NUREG-0661 and will replace the suppression chamber water temperature instruments presently listed in the T.S.

Another change to the T.S. is to reduce the minimum required flow for two LPCI pumps in the same loop. The reduced flow of 12000 gpm is a result of the installation of an orifice plate in the RHR return line to eliminate vibration in that line. A new containment cooling analysis was performed for this configuration, and it was determined that this flow rate produces a long-term suppression pool temperature well within that necessary for stable and complete steam condensation and for adequate RHR and core spray pumps net positive suction head.

The third change to the T.S. for this subject is to eliminate the requirement of placing the reactor in cold shutdown and performing a torus inspection after extended operation of relief valves above a suppression pool temperature of 130°F. Since the torus is being extensively upgraded, in accordance with NRC requirements, to withstand dynamic loading significantly beyond that originally expected, the torus inspection is not necessary. This has previously been approved for Browns Ferry unit 2 in amendment No. 85 and Browns Ferry unit 1 in amendment No. 91.

The final change to the T.S. on this subject is to revise the bases for the present limits on temperature of water in the torus. The present bases for suppression pool temperature limits were founded on the Humboldt Bay and Bodega Bay tests. Consistent with the long-term torus

integrity program of NUREG-0661 and NUREG-0783, the bases require change to account for steam mass fluxes through the safety/relief valve (S/RV) T-quenchers. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate residual heat removal (RHR) and core spray pump net positive suction head.

As noted above, the Commission ordered that the above torus modifications be implemented. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The changes to the T.S. place operability and calibration requirements on the new temperature monitoring system. Since these are new instruments, the surveillance requirements are not presently in the T.S. Thus, adding these restrictions and controls is encompassed by example (ii) provided by the Commission.

The change to reduce the flow for two LPCI pumps in the same loop and the change to eliminate the torus inspection requirement do not result in any reduction to the margin of safety or increase the probability or consequence of any accident. The changes are clearly encompassed by example (vi) of the guidance provided by the Commission.

The bases for the suppression pool temperature limits are also being changed to account for steam mass fluxes through the safety relief valve T-quenchers as required by NUREG-0661 and NUREG-0783. The proposed bases describe assurances of stable and complete condensation of steam discharged through the S/RVs and adequate RHR and core spray pump net

positive suction head. The changes are necessary administrative follow up action essential to the implementation of improvements required by the Commission. Modifying these restrictions is encompassed by example (ii) provided by the Commission.

3. Scram Discharge Instrument Volume

The SDVs and SDIVs are being modified to address inadequacies identified by the partial rod insertion event on Browns Ferry Unit 3 in June 1980. One of the modifications includes adding another valve in series to the existing drain and vent valves on the SDV and SDIV. Another modification includes adding electronic level switches to initiate a scram on a high level in the SDIV. On June 24, 1983, the Commission issued Orders for the Browns Ferry Nuclear Plant, Units 1 and 3 to install permanent Scram Discharge System modifications during the Cycle 5 outages for Units 1 and 3. (This is the Cycle 5 outage for Unit 3.) The modifications have been previously completed for Units 1 and 2. The Orders included "Model Technical Specifications which are provided as guidance for preparing Technical Specification changes that will be required to be approved before operation with the modified system." Both the modification of the systems and submission of T.S. changes to place operability and surveillance requirements on the new instruments and valves were required of the licensee to be in compliance with a Commission Order. Thus, the changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. Adding these new restrictions and controls, which otherwise would not be in the T.S., is encompassed by example (ii) of the guidance provided by the Commission.

4. Accident Monitoring Instrumentation

Item II.F.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires all licensees to install five new monitoring systems and to provide onsite sampling/analysis capability for a specified range of radionuclides. For all six categories, NUREG-0737 states: "Changes to technical specifications will be required." During this refueling outage, the licensee has installed: (a) a containment high-range monitoring system, (b) a drywell wide-range pressure monitoring system and (c) a suppression chamber wide-range water level monitoring system. These three items were required by NUREG-0737, items II.F.1.3, II.F.1.4 and II.F.1.5, respectively. The changes to the T.S., which track the model T.S. provided to the licensee by the staff, are to add operability and surveillance requirements on the new monitoring systems to the T.S.

The revisions also delete the present drywell pressure and suppression chamber water level instruments since they are being replaced by items b and c above. The changes to the technical specifications are necessary administrative follow up actions required by the Commission. Adding the new surveillance requirements and controls is encompassed by example (ii) of the guidance provided by the Commission.

5. Scram Permissive Pressure Switches at 1055 psig

Present configuration on unit 3 has a bypass function which allows a scram in the refuel and startup/hot standby modes of operation by the scram functions main steamline isolation valve closure and turbine condenser low vacuum when the reactor pressure is greater than 1055 psig.

The reactor high-pressure scram is set at 1055 psig and is operable in these two modes of operation. If reactor pressure exceeds 1055 psig, the reactor scrams due to the reactor high-pressure scram function, and the main steamline isolation valve closure and the turbine condenser low vacuum functions become operable. The bypass circuit therefore serves no real purpose. When the two scram functions become available, the reactor is already scrammed. Since the reactor is protected by the high-pressure scram function, the proposed change does not result in any reduction in the margin of safety. The T.S. changes therefore are encompassed by example (vi) of the guidance provided by the Commission.

6. H_2/O_2 Analyzer System

The isolation valves for the H_2/O_2 system are class B valves, those that isolate lines that connect directly with the containment free air space. This type of lines generally has two isolation valves in series, both on the outside of containment. Four valves on this system were previously installed inside containment. During this current outage, the four valves are being moved outboard of the containment and the T.S. are being revised to reflect this. Since the change is being made to correct an error and achieve consistency, it is encompassed by example (i) of the guidance provided by the Commission.

7. Drywell Temperature and Pressure

The drywell temperature and pressure surveillance instrumentation is being upgrade this outage to provide qualified, more reliable instrumentation. The T.S. are being revised to reflect new instrument numbers. The surveillance requirements remain unchanged. The changes to the technical specifications are necessary administrative follow up actions required by the Commission and are clearly encompassed by example (i) of the guidance provided by the Commission.

8. Testable Penetrations

Modifications are being made to the flange side of fourteen containment isolation valves which cannot be isolated from primary containment to be tested. This modification will provide two gaskets with a pressure tap between the gaskets to allow the flange to be leak tested. Operability of the valve will not be affected by this modification. Fourteen new testable penetrations resulted and they were added to the table of testable penetrations with double o-ring seals. New surveillance requirements are being added, the change is encompassed by example (ii) of the guidance provided by the Commission.

Several editorial changes were also made to this table. They include revising the identification name on several penetrations, adding two penetrations that have been tested but were inadvertently left out of the table and removing penetration X-213A which no longer exists. These changes are purely administrative and are encompassed by example (i) of the guidance provided by the Commission.

9. Redundant Air Supply to Drywell

During the current outage, TVA has installed a second discharge line from the drywell compressor into containment. This line was added to provide the capability for isolation of approximately one-half of the drywell suppression equipment in the case of a drywell line leak. This air supply will be used to supply two inboard main steam isolation valves (MSIVs), approximately one-half of the main steam relief valves (MSRVs), and approximately one-half of all other air-operated equipment in the drywell. This will significantly reduce the possibility of any one control air pipe break inside containment from requiring immediate shutdown and isolation due to MSIVs, MSRVs, and drywell coolers being inoperable. Since any line penetrating containment requires two isolation valves, the table in the Technical Specifications listing the isolation valves that must be periodically tested is being revised to add these two new isolation valves. TVA has concluded that this modification will increase the margin of safety. The changes to the technical specifications are necessary administrative follow up actions essential to the implementation of this improvement. The two isolation valves being added to the T.S. are new valves not presently listed in the T.S. If they were not added to the table of valves to be periodically tested, there would be no T.S. requirement to test these valves. Adding these additional controls is encompassed by example (ii) of the guidance provided by the Commission.

One isolation valve on the demineralized water system was removed from unit 3. The demineralized water system is no longer used. The isolation valve was removed and the line capped. The T.S. are being revised to remove this valve from the table of valves to be tested. The changes to

the technical specifications are necessary administrative follow up actions essential to the implementation of the improvement. The changes are clearly encompassed by example (i) provided by the Commission.

10. Monitoring of RPS Power Supply

By letter dated August 7, 1978, the Commission advised TVA that during review of Hatch Unit 2, the staff had identified certain deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system (RPS). Pursuant to 10 CFR 50.54(f), TVA was required to evaluate the RPS power supply for Browns Ferry 1, 2, and 3 in light of the information set forth in our letter. Based on the review of TVA's response, by letter dated September 24, 1980, the staff informed TVA (and most other BWRs) that "we have determined that modifications should be performed to provide fully redundant Class IE protection at the interface of non-Class IE power supplies and the RPS." The staff also advised TVA that "we have found that the conceptual design proposed by the General Electric Company and the installed modification on Hatch are acceptable solutions to our concern." By letter dated December 4, 1980, TVA committed to install the required modifications. By letters dated October 30, 1981 and July 28, 1982, NRC sent TVA model Technical Specifications for electric power monitoring of the RPS design and modification. During the current outage of Unit 3, the RPS is being modified to provide a fully redundant Class IE protection at the interface of the non-Class IE power supplies and the RPS. This will ensure that failure of a non-Class IE reactor protection power supply will not cause adverse interaction to the Class IE reactor protection system.

The Technical Specifications are being revised similar to the model T.S. provided to TVA to reflect the limiting conditions for operation and surveillance requirements associated with the RPS modifications. Page 41 is being modified to add a description of these sections in the bases. The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations and controls, which are presently not in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

11. Analog Trip System

The RPS, the primary containment isolation system (PCIS), and the core standby cooling systems (CSCS) use mechanical-type switches in the sensors that monitor plant process parameters. These mechanical-type switches are very subject to drift in the set-point as is evident from the many licensee event reports (LERs) that have been submitted reporting calibration drifts in these swtiches.

Advances in technology make it possible to replace the mechanical-type switches with a more accurate and more stable electronic transmitter/ electronic switch system. For several years, TVA has been planning to replace existing pressure switches that sense drywell and reactor pressures with analog loops and modify the reactor water level indication loops to improve the reliability, accuracy and response time of this instrumentation. The modification involves removing one device and substituting other devices to perform the same function. Changes in design bases, protective function, redundancy, trip point and logic are not involved. Similar modifications have been approved for other BWRs.

As described previously, most of the changes to the T.S. are administrative in nature (i.e., adding the specific number and types of sensor and adding notes to describe how testing is conducted). As such, they are encompassed by example (i) of the guidance provided by the Commission. The changes in surveillance requirements relate to example (ii) of the guidance provided by the Commission. Some of the surveillance intervals have been decreased as appropriate for each new instrument. However, the overall effect of the changes in technical specifications will be to increase the total surveillance requirements in support of a more reliable instrumentation system.

12. Thermal Power Monitor

During this outage, the licensee is installing a flow-biased simulated thermal power monitor. These monitors are installed on most all BWRs; the justification for these monitors is discussed in the "Bases" for the APRM settings in the BWR Standard Technical Specifications, NUREG-0123 (BWR/4, STS, Section 2.2.1, page B2-7). The monitors are installed to have the APRM flow-biased neutron flux signal respond to the thermal flux rather than the neutron flux by accounting for the approximately 6-second thermal time constant of the fuel. The addition of the thermal power monitor will prevent a flow-biased neutron flux scram when a transient-induced neutron flux spike occurs that is a short time duration and does not result in an instantaneous heat flux in excess of transient limits. Neutron flux is damped by approximately a 6-second fuel time constant. This feature will reduce the number of scrams due to small fast flux transients such as those which result from control valve and MSIV testing and small perturbations in water level and pressure.

A thermal power monitor was installed in Browns Ferry Units 1 and 2 during their last outage and approved by Amendment No. 91 to Facility Operating License No. DPR-33 issued December 1, 1983 and Amendment No. 85 to Facility Operating License No. DPR-52 issued March 11, 1983 respectively.

As identified previously, the changes to the T.S. are to add operability and functional test frequency requirements for this new trip system and to add a description of this new trip system in the "Bases." The changes to the T.S. are necessary administrative follow up actions essential to the implementation of these improvements. The additional limitations, restrictions and controls, which are not presently included in the T.S., are encompassed by example (ii) of the guidance provided by the Commission.

13. Administrative Changes

Several administrative changes are being made to the Technical Specifications. These include revising the Table of Contents to reflect the changes discussed above, and miscellaneous editorial changes. These changes are editorial in nature and have no safety significance. These changes are encompassed by example (i) cited by the Commission as an action not likely to pose a significant hazards consideration.

Since all of the changes to the T.S. given in the thirteen areas above are encompassed by an example in the guidance provided by the Commission of actions not likely to involve a significant hazards consideration, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.