ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATIONS REVISIONS

BROWNS FERRY NUCLEAR PLANT

UNIT 2

(TVA BFN TS 248)

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 378" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, 105 sec. delay timer and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, 105 sec. delay timer, 12 1/2 min. delay timer, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and 62)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	$\geq$ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps.
2	Instrument Channel - Reactor High Pressure (PIS-3-204A-D)	$\leq$ 1120 psig	A	1. Above trip setting trips recirculation pumps.
2	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	$\leq$ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level,. low reactor water level permissive, 105 sec. delay timer and CSS or RHR pump running, initiates ADS.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(16)	ADS Timer	105 sec ± 7	A	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level, high drywell pressure or high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
1(16)	ADS Timer (12 1/2 min.) (High Drywell Pressure Bypass Timer)	12 1/2 min. ± 2	A	1. Above trip setting, in conjunction with low reactor water level permissive, low reactor water level, 105 sec. delay timer, and RHR or CSS pumps running, initiates ADS.
2	Instrument Channel – RHR Discharge Pressure	100 ±10 psig	A	1. Below trip setting defers ADS actuation.
2	Instrument Channel CSS Pump Discharge Pressure	185 ±10 psig	A	1. Below trip setting defers ADS actuation.
1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid ±0.4	A	1. Alarm to detect core sparger pipe break.
	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

TABLE 4.2.B (Continued)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Pressure (PS-68-93 & 94)	(1)	Once/3 months	none
Core Spray Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	Once/operating cycle	none
ADS Timer (105 sec.)	(4)	Once/operating cycle	none
ADS Timer (12 1/2 min.) (High Drywell Pressure Bypass Timer)	(4)	Once/operating cycle	none

3.2 BASES (Cont'd)

The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and, in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus, the results given above are applicable here also.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the high drywell pressure bypass timer timed out (12 1/2 min.), and a 105 second time delay. In addition, at least one RHR pump or two core spray pumps must be running.

The high pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the bypass timer set at 15 minutes, a Peak Cladding Temperature (PCT) of 1424° F is reached for the worst case event. This temperature is well below the limiting PCT of 2200° F.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F, and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves.

3.2 BASES (Cont'd)

The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established nominal setting of three times normal background and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm with a nominal setpoint of 1.5 x normal full-power background is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below 825 psig.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

High temperature at the Reactor Cleanup System floor drain could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

3.2 BASES (Cont'd)

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted; however, the plant flood protection is always in place and does not depend in any way on advanced warning. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

The operability of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The operability of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for Browns Ferry Nuclear Plant. The instrumentation provided is consistent with specific portions of the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes."

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentration of potentially explosive gas mixtures in the offgas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20 Appendix B, Table II, Column 2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

4.2 BASES

The instrumentation listed in Tables 4.2.A through 4.2.F will be functionally tested and calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System of 0.99999 generally applies for all applications of (1-out-of-2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bistable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1-out-of-n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3}$$

$$= 40 \text{ days}$$

For additional margin a test interval of once per month will be used initially.

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

ENCLOSURE 2

DESCRIPTION AND JUSTIFICATION
BROWNS FERRY NUCLEAR PLANT
UNIT 2

Description of Change

The proposed amendment to the Browns Ferry Nuclear Plant Technical Specifications for Unit 2 would change the trip setpoint for the existing Automatic Depressurization System (ADS) timers from 120 ± 5 seconds to 105 ± 7 seconds and would add the surveillance and setpoint requirements for the high drywell pressure bypass 12 1/2 minute timers. These new timers are being added to bypass the high drywell pressure signal required for ADS initiation. This bypass timer meets the requirements of NUREG-0737, Item II.K.3.18. The changes involve updating and correcting all references to the present ADS timers in table 3.2.B, "Instrumentation that Initiates or Controls the Core and Containment Cooling Systems," pages 3.2/4.2-14, -15 and -17, as well as table 4.2.B, "Surveillance Requirements for Instrumentation That Initiate or Control The Core Standby Cooling System (CSCS)," page 3.2/4.2-45. A change to the Bases for the instrumentation section is also included. Refer to attached technical specification pages 3.2/4.2-14, -15, -17, -45, and -66 for detailed changes.

Reason for Change

The technical specifications for the ADS are being revised based on a modification to the initiating circuit to meet NUREG-0737, Item II.K.3.18, to update the setpoint of the present time-delay relays and to correct minor discrepancies. The tables affected, 3.2.B and 4.2.B, define the setpoint and surveillance requirements for the CSCS.

Justification for Change

The setpoint and tolerance values for the ADS time delay relays are a tradeoff between depressurizing the reactor vessel as quickly as possible but still not initiating ADS for short-lived transients and other events for which depressurization is not desired. The time delay must be long enough to allow the operator to determine the validity of the ADS initiation signal but still initiate ADS before the onset of any fuel damage.

The proposed ADS time delay (105 ± 7 seconds) includes the main steam safety relief valve (MSRV) stroke time and instrument error. The maximum time delay for ADS injection for the 105-second time-delay relay will now be: 105-seconds setpoint, + 7-seconds for technical specification tolerance and + 0.5-seconds MSRV opening time for a maximum time of 112.5-seconds. This is below the 120-second time limit assumed in the safety analysis (Final Safety Analysis Report (FSAR) Appendix N, Section N.6.5.10). The new technical specification span of 98 to 112-seconds is within the 90 to 120-seconds time span recommended in General Electric (GE) Service Information Letter No. 230 (June 6, 1977). The ADS logic will still allow the operator enough time to abort automatic depressurization by resetting the timers located in the control

Justification for Change (Cont'd)

room if conditions warrant such action. The reset restarts the timer if the requisite initiating conditions still exist. The 105-seconds setting for the timers is also consistent with the GE Standard Technical Specifications. The timer calibration tolerance was calculated to be ± 5.25-seconds but for simplicity it will be specified as ± 5-seconds. The ± 5.25-seconds tolerance ensures a high level of confidence that the timer will remain within the technical specification requirement of ± 7-seconds.

A high drywell pressure bypass timer is being added to meet the requirements of NUREG-0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. A calculation was performed by GE ("Bypass Timer Calculation for the ADS Modification for Browns Ferry Nuclear Plant," (DRF-A00-03088), dated May 1988). In that calculation, GE recommended that the time delay for the high-drywell pressure bypass timers be at least 10 but not more than 15 minutes for a Loss of Coolant Accident which does not result in high-drywell pressure. The worst case condition was a main steam line break outside primary containment. Time zero for the analysis was the point at which the reactor water level reached level 1. Level 1 is 378 inches above vessel zero. In the analysis, GE limited the maximum Peak Cladding Temperature (PCT) to 1500° F to assure that it would be bounded by the current plant safety analysis. The limiting condition of concern is a PCT of 2200° F. The analysis predicted that with the bypass timers set at 15 minutes, the PCT would be 1424° F. This temperature, which would occur at 20 minutes and 12-seconds into the event, is well below the limiting condition of a PCT of 2200° F.

The lower limit on the timer setting must be selected such that the Reactor Pressure Vessel (RPV) water level has been given sufficient time to recover to level 1 for a transient event or to give the unit operator sufficient time to determine if ADS is actually needed and take appropriate action. The GE analysis states that a minimum time delay of 10 minutes for the high drywell pressure bypass timers is sufficient to accomplish this goal.

The new high drywell pressure bypass timers will be set at 12 1/2 minutes which is midway between the GE recommended settings. The safety function of the ADS is to limit fuel barrier temperature in order to prevent uncontrolled release of fission products. The GE analysis shows that a timer setting of 12 1/2 minutes will allow the ADS to achieve that goal. Therefore, the ADS will be capable of performing its intended safety function.

The technical specification trip level setting entry for the new 12 1/2 minute ADS time delay relays will be "12 1/2 minutes ± 2 minutes." The maximum allowable time delay for the bypass timer is 14 1/2 minutes, 12 1/2 minutes + 2 minutes, which is below the maximum value of 15 minutes recommended by GE for the additional timer.

The minimum allowable time delay for the bypass timer is 10 1/2 minutes, which is above the minimum of 10 minutes recommended by GE. The timer calibration tolerance was calculated to be ± 86.5 seconds, but for simplicity it will be specified as ± 86 seconds. The 86-second tolerance ensures a high level of confidence that the timer will remain within the technical specification requirement of ± 2 minutes (120 seconds).

The wiring and components for the circuit modification are similar to or better than those presently used so reliability of the new circuits is the same as or better than the present circuits. Both the present (105 second) and the new (12 1/2 minute) timers are Series 7000 Agastat Time Delay Relays so their characteristics will be the same. The possibility of spurious actuation due to equipment failure, testing, or maintenance errors is slightly greater for the new circuit than for the present one because the modifications provide more paths to complete the ADS initiation (Boiling Water Reactor Owners Group's (BWROG) Evaluation of NUREG-0737, Item II.K.3.18, "Modification of ADS Logic," submitted to NRC under a cover letter dated October 28, 1982, BWROG-8260, page 57). However, the possibility of an unnecessary RPV depressurization is not significantly affected by the proposed modification because any slight increase in the chance of spurious actuation is offset by a decrease in the possibility of inadvertent manual depressurization due to operator error (BWROG, Evaluation of NUREG-0737, Item II.K.3.18, page 56).

Additionally, administrative changes are made to the tables to avoid confusion between the two different timers. On page 3.2/4.2-15, "drywell high pressure," is being changed to "low reactor water level permissive" and on page 3.2/4.2-17, "Low Pressure Coolant Injection (LPCI)," is being changed to "Residual Heat Removal(RHR)" to correct errors and provide consistency in the Remarks section of table 3.2.B. These changes are administrative in nature and will not affect the intent of any technical specification.

Entries for the new 12 1/2 minute timers are consistent with the entries for the present ADS timers. With these additions, the setpoint requirements and the surveillance requirements for the two types of ADS timers will be the same.

The Bases for section 3.2 are updated to reflect the initiation signals required to automatically start ADS. The reason for the new ADS timer and the Bases for its setpoint are included.

ENCLOSURE 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION
BROWNS FERRY NUCLEAR PLANT
UNIT 2

Description of Amendment Request

The proposed amendment to the Browns Ferry Nuclear Plant, Unit 2, Technical Specification would change the trip setpoint for the existing Automatic Depressurization System (ADS) timers from 120 ± 5 seconds to 105 ± 7 seconds, correct minor discrepancies, and would add the surveillance and setpoint requirements for the new 12 1/2 minute timers being added in accordance with NUREG-0737, Item II.K.3.18, to bypass the high drywell pressure requirement for ADS initiation. The proposed amendment would revise technical specifications table 3.2.B, "Instrumentation that Initiates or Controls the Core and Containment Cooling Systems," table 4.2.B, "Surveillance Requirements for Instrumentation that Initiate or Control Core Standby Cooling Systems," which contain the setpoint and surveillance requirements for instruments involved in the ADS, and the bases for the insturmentation section.

Basis for Proposed No Significant Hazards Consideration Determination

NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from an accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because:

    (a) None of the Final Safety Analysis Report (FSAR) accidents are initiated by spurious actuation or failure of the ADS initiation logic circuitry.

    (b) The ADS will still function and operate as currently designed, the 120 ± 5 second setpoint for the existing timer is still long enough to allow the operator sufficient time to abort an ADS blowdown if conditions warrant, (e.g., if high pressure makeup systems are adequately restoring Reactor Pressure Vessel (RPV) water level). Thus, lowering the setpoint of the existing ADS timer to 105 ± 7 seconds does not significantly increase the probability of creating the large stresses associated with a rapid blowdown event on the RPV.

These large stresses caused by a rapid blowdown, however, were included in the design of the reactor vessel, and these stresses will not be increased by the proposed changes.

(c) With this modification, two means (manually by operator and automatically by new ADS circuitry) would be available to mitigate a Loss of Coolant Accident event. Therefore, the consequences of this accident scenario are reduced.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The modification to decrease the setpoint ensures that the ADS reacts to an accident condition within the time analyzed in the FSAR. The addition of a timer to bypass the high-drywell pressure interlock causes the ADS to initiate for accidents which previously required manual initiation and decreases the probability for operator error.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The time required for high pressure reactor vessel makeup systems to react to a loss-of-coolant accident is reduced (i.e., 120-second timer setpoint changed to 105-second setpoint), but it is still long enough to ensure that short term transient conditions will not unnecessarily initiate the ADS. Also, by adding circuity to automatically initiate ADS for accidents previously requiring manual initiation, it decreases the chance for the operator error.

Determination of Basis for Proposed No Significant Hazards

Since the application for amendment involves a proposed change that is encompassed by the criteria for which no significant hazards consideration exists, TVA has made a proposed determination that the application involves no significant hazards consideration.