January 30, 1989

Docket No. 50-260

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Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) TIMER - NUREG 0737, ITEM II.K.3.18 LICENSE AMENDMENT TO CHANGE TECHNICAL SPECIFICATION BROWNS FERRY NUCLEAR PLANT, UNIT 2 (TAC 00090) (TS 248)

The Commission has issued the enclosed Amendment No. 162, to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. This amendment is in response to your application dated July 29, 1988.

It is our understanding that TVA plans to incorporate surveillance of the manual inhibit switch in a BFN on-site surveillance procedure consistent with the BWR Standard Technical Specifications, NUREG-0123. We find this acceptable.

It is our further understanding that BFN will comply with the final recommendations of the Technical Specification Improvement Program regarding this issue.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Original signed by

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 162 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

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*SEE PREVIOUS CONCURRENCE

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NAME	:MSimms	:DMoran	:	:SBlack	:	:	:
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Docket No. 50-260

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See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 29, 1988, 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 Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162 , 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 its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 30, 1989

3.2 <u>BASES</u>

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at 17.7" (378" above vessel zero) above the active fuel (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

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.3.18. 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.

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 (RWCU) System floor drain in the space near the RWCU system or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

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:

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

$$= 40 \text{ days}$$

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

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

(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.

Forbidding simultaneous testing improves the availability of the system over that whic ould be achieved by testing ea channel independently. These one-out-of-n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by Curve No. 1 of Figure 4.2-1 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^3 hours.

If two similar channels are used in a 1-out-of-2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

MODIFICATION TO THE AUTOMATIC DEPRESSURIZATION SYSTEM LOGIC

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

The final recommendation of the NRC's Bulletin and Orders Task Force, to mitigate accidents, required applicants for boiling water reactor (BWR) plants to perform a feasibility and risk assessment study. This study would determine the optimum Automatic Depressurization System (ADS) design modification that would eliminate the need for manual actuation to ensure adequate core cooling during certain accident and transient conditions with a loss of reactor coolant and no high drywell pressure.

In response to the TMI requirement, issue II.K.3.18 of NUREG-0737, concerning the ADS system, the BWR Owners Group (BWROG) conducted a study and identified modifications that would satisfy the basic concerns of manual versus automatic actuation of the ADS system. A study of alternatives was presented to the NRC staff by BWROG for ADS logic modifications to eliminate manual actuation and ensure core coverage.

By letter dated June 3, 1983, the NRC staff transmitted to TVA an evaluation of the BWROG study applicable to the Browns Ferry nuclear plants. The BWROG study contained seven alternatives to the present ADS logic. The NRC staff evaluation of the study concluded that the following two alternatives are acceptable for ADS logic modification:

a. Elimination of the high drywell pressure permissive and addition of a manual inhibit switch.

b. Bypass of the high drywell pressure permissive after a sustained low water level and addition of a manual inhibit switch.

The NRC staff concluded that for both options listed, the manual inhibit switch should be addressed by TVA in the emergency operating procedures, and a surveillance plan for the switch should be included in the Technical Specifications. Also, for adoption of the second option, the setting of the bypass timer should be justified, and a plan for periodic testing of the timer should be submitted by TVA.

A meeting was held between TVA and the NRC staff on July 14, 1986, to close the issue. Subsequently, TVA elected to implement a modification similar to option b described above. The ADS logic would be modified to allow the ADS to initiate vessel depressurization, automatically bypassing the high drywell pressure signal 10 minutes after a sustained lo-lo-lo reactor vessel water level signal. The 10-minute delay would consist of a new 8-minute bypass timer in conjunction with the existing ADS 2-minute timer. A manual switch would also be installed to allow the operator to inhibit the ADS as recommended by the BWROG emergency procedure guidelines. The above modification proposed by TVA during the July 14, 1986 meeting was confirmed by TVA in their letter to the NRC staff on March 5, 1987.

On July 29, 1988, TVA submitted, by letter, a request for amendment to Operating License DPR-52 to change the Technical Specifications (TS) for Unit 2 to change the trip setpoint for the existing ADS timer and add the surveillance and setpoint requirements for a high drywell pressure bypass timer.

On August 10, 1988, the NRC staff requested, by letter, that TVA provide additional information on the proposed ADS modification. The staff requested that the complete design package change for the ADS actuation modification be provided, including TVA's safety evaluation and all drawings. This additional information was submitted to the NRC staff by TVA in a letter dated September 12, 1988. This information did not change the substance of the originally proposed action which was noticed in the Federal Register on August 24, 1988 (53 FR 32297). Also, it did not affect the staff's initial determination made in that notice.

2.0 EVALUATION

The ADS depressurizes the reactor vessel so that the low-pressure emergency core cooling system can inject water into the reactor vessel following small or intermediate size loss-of-coolant-accidents (LOCA) concurrent with the high-pressure coolant injection (HPCI) system failure. The ADS system consists of redundant signal logic arranged in two separate solenoid-operated air pilot valves on the six main steam relief valves that relieve steam to the suppression pool. The two separate logic buses, A and B, are comprised of two channels for each logic bus.

The ADS is actuated when the following conditions exist:

a. Low reactor water level permissive (level 3) 378" above vessel zero
Instrument Channels LIS-3-58 A, B, C, and D

b. Low reactor water level (level 1) 544" above vessel zero
Instrument Channels LIS-3-184 and 185

c. Drywell high pressure, 2.5 PSIG
Instrument Channels PSI-64-57 A, B, C, and D

d. Time delay 120 seconds has expired
Timers 2E-K34 and 2E-K35

e. One low-pressure coolant injection (LPCI)-RHR pump running or two core spray pumps running

LPCI-RHR pump discharge pressure PS-10-123 A, B, C, and D
Core spray pump discharge pressure PS-14-44 A, B, C, D

The following ADS design modifications were identified in TVA's proposed amendment to Technical Specification No. 248 and Design Change Request 3478 to comply with NUREG-0737, Item II.K.3.18, and BWROG recommendations.

f. Replacement of time delay relays, item d above, with qualified timers set for 105 seconds, plus or minus 7 seconds.

g. Addition of four time delay pick-up relays to bypass, item c above, high drywell pressure. The timers are set for 12 1/2 minutes, plus or minus 2 minutes.

h. Addition of two keylock control switches for each logic bus A and B to allow the operator to inhibit (block) the ADS initiation logic.

i. Addition of annunicator window number 31, annunicator XX-55-3C, control panel 9-3, to indicate to the operator that either logic bus A or B is inhibited.

The replacement of the existing ADS timer relay, item d above, with a Class 1E qualified timer relay, item f above, and changing the setpoint from 120 seconds, plus or minus 5 seconds, to 105 seconds, plus or minus 7 seconds, improve safety by ensuring ADS initiation within the time required. This required time is stated in Browns Ferry Nuclear (BFN) Final Safety Analysis Report (FSAR), Appendix N, Section N.6.5.10. The FSAR states that a time delay of 120 seconds is the maximum practical delay time for ADS initiation.

The installation of the high drywell pressure bypass timer, item g above, provides a means to ensure adequate core cooling for a main steam pipe break accident outside the primary containment with the HPCI inoperable. The time delay setting for the bypass timer allows for:

a. Avoidance of excess fuel cladding heatup.

b. Sufficient time to allow high-pressure makeup systems to recover the reactor vessel water level.

c. Sufficient time for the operator to prevent ADS from occurring during an anticipated transient without scram (ATWS) event.

TVA's letter of September 12, 1988, to the staff provided an analysis of the effects on safety of the ADS design modification. This analysis defined the design basis for the bypass timer to be the main steam pipe break outside the primary containment accident. The analysis defined the analytical limits for the ADS bypass timer as 15 minutes to 10 minutes, based on the General Electrical Analysis DFR-A00-03088.

TS change No. 248 for the ADS submitted by TVA on July 29, 1988, revised Section 3.2, bases of the TS. This section states that for the worst-case condition, steam line break outside the primary containment with HPCI inoperable and the timer set at 15 minutes, a peak cladding temperature (PCT) of 1424°F will be reached. This temperature is less than the limiting PCT of 2200°F. This analysis for PCT versus time delay was also stated in the General Electric analysis previously mentioned.

The bypass timer will be functionally tested during the ADS logic system functional test which is performed once each operating cycle. The manual ADS inhibit switch and the associated annunication have not been addressed by TVA relative to the TS limiting condition for operation (LCO) and functional surveillance of the switch.

TVA has revised Surveillance Procedure BF SI-4.2.B-44, Revision 2, to address the bypass timer functional test and Emergency Operating Procedure, Appendix 3, EOI 1, Revision 0, to address the operator use of the manual inhibit switch.

By letter dated May 9, 1988, the NRC Director of the Office of Nuclear Reactor Regulations transmitted an evaluation of the BWR Owners Group (BWROG) report identifying which Standard Technical Specification (STS) requirements BWROG believes should be retained in the new STS and which can be relocated in other licensee-controlled documents. In this evaluation, the BWR-Table 1, Section 3.3.3, Report Item 112B, the ADS manual inhibit switch is listed under LCOs to be retained. It is our understanding that TVA plans to incorporate surveillance of the manual inhibit switch in a BFN on-site surveillance procedure consistent with the BWR Standard Technical Specification, NUREG-0123.

It is our further understanding that BFN will comply with the final recommendations of the Technical Specification Improvement Program regarding this issue.

3.0 ACCEPTANCE

The staff concludes that the proposed TVA design modification and TS change meet the requirements of Item II.K.3.18 of NUREG-0737. The following specific items are acceptable to the staff:

a. TVA has provided a basis for the setpoint of the high drywell bypass timer and has addressed the timer functional test surveillance.

b. TVA has provided a manual ADS inhibit (block) switch as recommended by BWROG.

c. The ADS modification does not decrease the safety function of the ADS system.

It is our understanding that TVA plans to incorporate surveillance of the manual inhibit switch in a BFN on-site surveillance procedure consistent with the BWR Standard Technical Specification, NUREG 0123.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 32297) on August 24, 1988 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

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Dated: January 30, 1989