CENSE AUTHORITY FILE COPY

DO NOT REMOVE

May 13, 1987

Docket Nos. 50-259, 260, 296

Posted
Amdt. 129
to DPR-52

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

SUBJECT: ROD WORTH MINIMIZER AND ROD SEQUENCE CONTROL SYSTEM REQUIREMENTS (TAC 61960/61961/61962)

Re: Browns Ferry Nuclear Plant, Units 1, 2 and 3

The Commission has issued the enclosed Amendments Nos. 133, 129, and 104 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3. These amendments are in response to your application dated June 4, 1986 (TVA BFNP TS 220).

The amendments change the Technical Specifications relating to requirements for the Rod Worth Minimizer and the Rod Sequence Control System.

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

You should assure that any safety-related employee concerns presently under review by the Employee Concern Task Group and pertaining to this issue are appropriately addressed prior to the start-up of the Browns Ferry Units.

Sincerely,

Original signed by:

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Enclosures:
1. Amendment No. 133 to License No. DPR-33
2. Amendment No. 129 to License No. DPR-52
3. Amendment No. 104 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY

DISTRIBUTION:
Docket File
NRC PDR
Local PDR
JAxelrad
SEbneter
SRichardson
JZwolinski
JStang
GGears
CJamerson (4)
DHagan
EButcher
Wanda Jones
BDLiaw
GZech, RII
SRConnelly
BHayes
FMiraglia
TBarnhart (12)
ACRS (10)
JPartlow
OGC-BETH
ARM/LFMB
GPA/PA
EJordan
Projects Rdg
BFN File

*See Previous Concurrence

DBL:PD#2*	DBL:PD#2*	DBL:PD#2*	OGC - Bethesda*	DTVA:AD/P
SNorris:	GGears:jch	MGrotenhuis	MYoung	JZwolinski
2/4/87	2/4/87	2/4/87	2/18/87	5/13/87

g 5/1/87

Mr. S. A. White
Tennessee Valley Authority

Browns Ferry Nuclear Plant
Units 1, 2, and 3

cc:
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 330
Knoxville, Tennessee 37902

Mr. R. W. Cantrell
ATTN: D. L. Williams
Tennessee Valley Authority
400 West Summit Hill Drive, W12 A12
Knoxville, Tennessee 37902

Mr. R. L. Gridley
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. M. J. May
Tennessee Valley Authority
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

Mr. H. P. Pomrehn
Tennessee Valley Authority
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

Chairman, Limestone County Commission
P.O. Box 188
Athens, Alabama 35611

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. Steven Roessler
U. S. Nuclear Regulatory Commission
Reactor Training Center
Osborne Office Center, Suite 200
Chattanooga, Tennessee 37411

Resident Inspector/Browns Ferry NP
U.S. Nuclear Regulatory Commission
Route 2, Box 311
Athens, Alabama 35611

Mr. Richard King
c/o U.S. GAO
1111 North Shore
Suite 225, Box 194
Knoxville, Tennessee 37919

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. 129
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 June 4, 1986, 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. 129, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Assistant Director
for Projects
Division of TVA Projects
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 13, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the areas of changes. Overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
i	i
ii	ii
3.3/4.3-5	3.3/4.3-5*
3.3/4.3-6	3.3/4.3-6
3.3/4.3-7	3.3/4.3-7
3.3/4.3-8	3.3/4.3-8
3.3/4.3-9	3.3/4.3-9
3.3/4.3-10	3.3/4.3-10**
3.3/4.3-11	3.3/4.3-11**
Bases 3.3/4.3-12	3.3/4.3-12**
Bases 3.3/4.3-13	Bases 3.3/4.3-13**
Bases 3.3/4.3-14	Bases 3.3/4.3-14**
Bases 3.3/4.3-15	Bases 3.3/4.3-15**
Bases 3.3/4.3-16	Bases 3.3/4.3-16**
Bases 3.3/4.3-17	Bases 3.3/4.3-17**
Bases 3.3/4.3-18	Bases 3.3/4.3-18**
Bases 3.3/4.3-19	Bases 3.3/4.3-19**
--	Bases 3.3/4.3-20**

**Pagination change only

TABLE OF CONTENTS

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the REFUEL condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. Verify that the control rod is following the drive by observing a response in the nuclear instrumentation each time a rod is moved when the reactor is operating above the preset power level of the RSCS.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.a Whenever the reactor is in the startup or run modes below 20% rated power, the Rod Sequence Control System (RSCS) shall be OPERABLE, except that the RSCS constraints may be suspended by means of the individual rod bypass switches for
1 - special criticality tests, or

2 - control rod scram timing per 4.3.C.1.

When RSCS is bypassed on individual rods for these exceptions, RWM must be operable per 3.3.B.3.b and a second party verification may not be used in lieu of RWM.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.a.1 The Rod Sequence Control System (RSCS) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. Performance of the comparator check of group notch circuits within 8 hours prior to control rod withdrawal for the purpose of making the reactor critical

b. Selecting and attempting to withdraw an out-of-sequence control rod after withdrawal of the first insequence control rod.

c. Attempting to withdraw a control rod more than one notch prior to other control rod movement after the group notch mode is automatically initiated.

3.a.2 The Rod Sequence Control System (RCS) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. Performance of the comparator check of the group notch circuits within 8 hours prior to automatic initiation of the group notch mode.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.B. Control Rods	4.3.B. Control Rods

4.3.B. Control Rods

3.a.2 (Cont'd)

b. Attempting to insert a control rod more than one notch prior to other control rod movement after the group notch mode is automatically initiated.

c. Selecting and attempting to move an out-of-sequence control rod after insertion of the first insequence control rod after reaching a black and white rod pattern.

3.3.B. Control Rods

3.b. Whenever the reactor is in the startup or run modes below 20% rated power, the Rod Worth Minimizer (RWM) shall be OPERABLE. With the RWM INOPERABLE, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the plant staff who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.

4.3.B. Control Rods

3.b.1 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor startup by the following checks:

a. By demonstrating that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical verify proper annunciation of the selection error of at least one out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.1 (Cont'd)

c. Within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

3.b.2 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. By demonstrating that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to RWM automatic initiation when reducing thermal power, verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within one hour after RWM automatic initiation when reducing thermal power, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

3.3/4.3 REACTIVITY CONTROL

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.c. If Specifications 3.3.B.3.a through .b cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:

   a. Both RBM channels shall be operable:

   or

   b. Control rod withdrawal shall be blocked.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.3 When the RWM is not OPERABLE a second licensed operator or other technically qualified member of the plant staff shall verify that the correct rod program is followed except as specified in 3.3.B.3.a.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and at least once per 24 hours thereafter.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all OPERABLE control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.500

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

1. After each refueling outage, all OPERABLE rods shall be scram-time tested from the fully withdrawn position with the nuclear system pressure above 800 psig. This testing shall be completed prior to exceeding 40% power. Below 20% power, only rods in those sequences (A_{12} and A_{34} or B_{12} and B_{34}) which were fully withdrawn in the region from 100% rod density to 50% rod density shall be scram-time tested. The sequence restraints imposed upon the control rods in the 100-50 percent rod density groups to the preset power level may be removed by use of the individual bypass switches associated with those control rods which are fully or partially withdrawn and are not within the 100-50 percent rod density groups. In order to bypass a rod, the actual rod axial position must be known; and the rod must be in the correct in-sequence position.

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

a. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the shutdown condition within 24 hours.	4.3.E. Surveillance requirements are as specified in 4.3.C and .D above.
F. Scram Discharge Volume (SDV)	F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open prior to each STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.
	1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE monthly.
2. In the event any SDV drain or vent valve becomes INOPERABLE, reactor operation may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is INOPERABLE, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become INOPERABLE, the reactor shall be in hot standby within 24 hours.	3. No additional surveillance required.

3.3/4.3 BASES

A. Reactivity Limitation

1. The requirements for the control rod drive system have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.4 of the Final Safety Analysis Report, the control rod system design is intended to provide sufficient control of core reactivity that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 percent Wk with the analytically determined strongest control rod fully withdrawn.

   The value of "R", in units of percent Wk, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

   The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this "analytically strongest" rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of the control cell geometry and local k_{χ}. Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the "analytically strongest") is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38 percent Wk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity Margin - INOPERABLE Control Rods - Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and disarmed electrically*, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a nonfully inserted position, that position shall be consistent with the shutdown reactivity limitations stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest OPERABLE control rod does not insert. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress-assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed rod after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings. The Rod Sequence Control System is not automatically bypassed until reactor power is above 20 percent power. Therefore, control rod movement is restricted and the single notch exercise surveillance test is only performed above this power level. The Rod Sequence Control System prevents movement of out-of-sequence rods unless power is above 20 percent.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Sequence Control System and the rod worth minimizer.

* To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in subsection 3.5.2 of the FSAR and the safety evaluation is given in subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to prespecified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Reference Sections 3.6.6, 7.7.A, 7.16.5.3, and 14.6.2 of the FSAR, and NEDO-10527 and supplements thereto.

   In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20 percent of rated. Material in the cited reference shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20 percent, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize individual control rod worth.

At power levels below 20 percent of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.16.5.3 of the FSAR. They serve as a backup to procedure control of control rod sequences, which limit the maximum reactivity worth of control rods. Except during specified exceptions, In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20 percent, these devices force adherence to acceptable rod patterns. Above 20 percent of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20 percent of rated power are imposed by power distribution requirements, as defined in Sections 3.5.I, 3.5.J, 4.5.I, and 4.5.J of these technical specifications. Power level for automatic bypass of the RSCS function is sensed by first stage turbine pressure.

Because it is allowable to bypass certain rods in the RSCS during scram time testing below 20 percent of rated power in the STARTUP or RUN modes, a second licensed operator is not an acceptable substitute for the RWM during this testing.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One OPERABLE SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two OPERABLE SRMs are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (i.e., MCPR given by Specification 3.5.k or LHGR of 13.4 kW/ft. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the nuclear engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of INOPERABLE control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.

C. Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during STARTUP of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and STARTUP testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10 percent of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWRs with control rod drives the same as those on Browns Ferry Nuclear Plant.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of INOPERABLE rods.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of control rod motion.

This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from sensor and circuit delays after which the pilot scram solenoid deenergizes to 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds, rather than 120 milliseconds, are conservatively assumed for this time interval in the transient analyses and are also included in the allowable scram insertion times of Specification 3.3.C.

In order to perform scram testing as required by Specification 4.3.C.1, the relaxation of certain restraints in the rod sequence control system is required. Individual rod bypass switches may be used as described in Specification 4.3.C.1.

The position of any rod bypassed must be known to be in accordance with rod withdrawal sequence. Bypassing of rods in the manner described in Specification 4.3.C.1 will allow the subsequent withdrawal of any rod scrammed in the 100 percent to 50 percent rod density groups; however, it will maintain group notch control over all rods in the 50 percent density to preset power level range. In addition, RSCS will prevent movement of rods in the 50 percent density to preset power level range until the scrammed rod has been withdrawn.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1 percent WK. Deviations in core reactivity greater than 1 percent WK are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. No BASES provided for this specification

F. Scram Discharge Volume

The nominal stroke time for the scram discharge volume vent and drain valves is $\leq$ 30 seconds following a scram. The purpose of these valves is to limit the quantity of reactor water discharged after a scram and no direct safety function is performed. The surveillance for the valves assures that system drainage is not impeded by a valve which fails to open and that the valves are OPERABLE and capable of closing upon a scram.

References

1. Generic Reload Fuel Application,
Licensing Topical Report, NEDE-24011-P-A and Addenda.

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

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKETS NOS. 50-259, 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated June 4, 1986 (Reference 1) Tennessee Valley Authority (TVA) proposed to change the Technical Specifications for the Browns Ferry Nuclear Plant Units 1, 2 and 3. The specifications to be changed are concerned with requirements for the Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS).

The current specifications (3/4.3.B.3) are in somewhat different format and have slightly different content for the three reactors. They also differ somewhat from Standard Technical Specifications. They were involved in a "Notice of Violation and Proposed Imposition of Civil Penalties" (Reference 2) in 1985. A subsequent letter (Reference 3) by the NRC suggested clarification of these specifications. Thus the intent of the changes is to provide (1) the same specifications for the three reactors, (2) clarification (and other editorial changes) of the specifications to avoid difficulties of interpretation, and (3) a closer approach to the Standard Technical Specifications.

2.0 EVALUATION

The current RSCS and RWM Limiting Conditions for Operation and Surveillance specifications are 3/4.B.3.a through d. In the proposed change they will be 3.3.B.3.a through c and 4.3.B.3.a.1 and 2 and b.1 through 3. Unlike the current specifications the numbering and wording of the new specifications will be the same for each of the three reactors. Since the systems and operations are the same for these reactors, this change will avoid confusion with respect to the commonality of operating requirements.

The changes to the specifications do not change the overall meaning and intent and significant requirements of the specifications. All of the changes are such as to move closer to (and in most respects directly adopt) the language and content of the Standard Technical Specifications for RCSC and RWM operation and surveillance. The revised specifications provide all of the operational limits and action and surveillance requirements of the Standard Technical Specifications (STS) and are clearly within all acceptable criteria with respect to the operation of the RCSC and RWM systems and components specified in Standard Review Plan (SRP 7.7). This is done both for reactor startup and shutdown operations. This change to provide the same requirements and much of the language of the STS is a satisfactory approach to needed modification and improvement of these specifications. Therefore, the changes proposed by TVA are acceptable.

The specific request by the NRC cited in Reference 3 was to clarify the specifications to indicate that when the RWM (or RSCS) is inoperable (below 20 percent power) the reactor need not be shut down, but can continue to operate if there is no motion of any control rod (except by scram). This is accomplished in the revised specifications by specification 3.3.B.3.b, in which no control rod movement is an acceptable alternative to immediate shutdown. This change is acceptable and satisfies the request of Reference 3.

TVA has proposed changes to the Technical Specifications of RSCS and RWM operation and surveillance which would make them the same for each of the three reactors, would adopt the requirements for operation, action and surveillance of the Standard Technical Specifications and therefore, clearly within all acceptable criteria of SRP 7.7 and would satisfy the NRC request for clarifications of the specifications. We have reviewed the changes and information submitted by TVA and based on this review we have concluded that appropriate material was submitted and that the proposed changes satisfy staff positions and requirements in these areas. Operations in the proposed manner and the Technical Specification changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 should be no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

We have 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 or to the health and safety of the public.

5.0 REFERENCES

1. Letter (and attachment) from R. Gridley, TVA, to D. Muller, NRC, dated June 4, 1986, "Proposed Technical Specification Revisions, Browns Ferry Nuclear Plants Unit 1, 2 and 3."

2. Letter from J. Grace, NRC, to H. Parris, TVA, dated February 27, 1985, "Notice of Violation and Proposed Imposition of Civil Penalties: EA 84-136 ...."

3. Letter from J. Taylor, NRC to H. Parris, TVA, dated August 05, 1985.

Principal Contributor: H. Richings

Dated: May 13, 1987