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Docket No. 50-260

6-18-76

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chatanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 20 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit No. 2. This amendment is in response to your request dated May 28, 1976 as supplemented by your submittal dated June 1, 1976.

The amendment revises the Technical Specifications to allow loading the fuel of Unit 2 in the Unit 2 reactor vessel. Operation is not authorized by this amendment.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures
1. Amendment No 20 to DPR-52
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:

See next page

| | | | | | | |
|---|---|---|---|---|---|---|
| x27433 OFFICE➤ | ORB#1 | OELD | ORB#1 AS | | KRG ADOR | ADSS |
| SURNAME➤ | TWambach:tsb | Drossman | ASchwencer | 6/18/76 | K.R. Goller | Tedesco for RL Tedesco |
| DATE➤ | 6/18/76 | 6/18/76 | 6/18/76 | | 6/18/76 | 6/18/76 |

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Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
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Montgomery, Alabama 36104

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
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 May 28, 1976 as supplemented June 1, 1976, 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 amendmed by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 18 1976

ATTACHMENT T◡ ˙ICENSE AMENDMENT

AMENDMENT NO. 20 TO DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

Replace the following pages with the enclosed pages.

| | | | | |
|---|---|---|---|---|
| 15 | 100 | 131 | 134 | 206 |
| 33 | 109 | 131a | 146 | 253 |
| 54 | 115 | 132 | 147 | 252 |
| 59 | 122 | 133 | 149 | 259 |
| 81 | 130 | 133a | 176 | 268 |

Delete page 256a and add pages 260a thru o, and 288a.

## 1.1/2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The objective of this specification is to assure that irradiated fuel in the reactor vessel remains covered with water at all times. When plant equipment as specified in facility technical specification is available, this objective is accomplished automatically. This is no longer possible. Accordingly, procedural control of the water level is required as set forth in the specification 3.2.A.1.

With the reactor in cold shutdown, rapid makeup of coolant inventory is not required. The provisions for manual operation under procedural controls provided by the TVA Safety Analysis of the BFNP units 1 and 2 included as Part VI, Section E, of the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" will assure timely isolation of leaks by requiring redundant level alarms and indicators. Double isolation features, one of which shall be operative from the control room, will prevent draining of the vessel as a result of any single active component failure. Any valves or other components required to be disabled by the SAR referenced above, will be disabled by opening power supply breakers and/or removing fuses. Also, in the unlikely event that coolant inventory is required, a core spray pump, capable of delivering flow, will be available to provide the required vessel makeup capability.

The limiting safety systems scram setpoints are not capable of performing their intended function since the automatic actions they would initiate are unavailable. However, the facility has been placed in a condition that prevents the need for these actions.

TABLE 3.1.A/4.1.A

REACTOR PROTECTION SYSTEM INSTRUMENTATION

| Minimum Number of Functional Channels per Trip System | Trip Function | Trip Level Setting | Functional Test Frequency | Alarm Level Setting | Calibration Frequency |
|---|---|---|---|---|---|
| 1 | Manual scram | N/A | once/month | N/A | N/A |
| 2 (1) | SRM high-high count | $\leq 5 \times 10^5$ cps | once/week | $\leq 5 \times 10^5$ cps (upscale HI HI)<br>$\leq 10^5$ cps (upscale HI)<br>$\geq 3$ cps (downscale) | once/3 mos |
| 2 (2) | High Water Level in Scram Discharge Tank | N/A | once/3 months | $\leq$ 50 Gallons | once/3 month |

Notes for Table 3.1.A/4.1.A

(1) The SRM's are presently connected in the non-coincident mode in the RPS, where any 1 SRM upscale HI HI will initiate a scram. Of the 4 SRM channels, only 2 are required to be functional.

(2) Channels A or C and B or D required.

Table 3.2.A

SURVEILLANCE INSTRUMENTATION

| Minimum # of Operable Instrument Channels | Instrument # | Instrument | Type Indication and Range | Alarm Setting | Notes |
|---|---|---|---|---|---|
| 2 | LI-3-206 or<br>LR-3-53 or<br>LI-3-53 or<br>LI-3-55 and<br>LI-3-46A or 46B | Reactor Water Level | Indicator 0" to 60"<br>Recorder 0" to 60"<br>Indicator 0" to 60"<br>Indicator 0" to 400"<br>Indicator +60" to -155" | Low $\geq$ 27", high $\leq$39" | (1)(4) |
| 2 | PI-3-54<br>PR-3-53 | Reactor Pressure | Indicator 0-1200 psig<br>Recorder 0-1200 psig | High $\leq$ 1040 psig | (1)(5) |
| 2 | PR-64-50 and<br>PI-64-67 | Drywell Pressure | Recorder 0-80 psig<br>Indicator 0-80 psig | | (1)(5) |
| 2 | TI-64-52A and<br>TR-64-52 | Drywell Temperature | Indicator 0-400° F.<br>Recorder 0-400° F. | High $\leq$ 145° F. | (1)(5) |
| 2 | TI-64-55A and<br>TIS-64-55 | Suppression Chamber Water Temperature | Indicators 0-400° F. | High $\leq$ 90° F | (1)(4) |
| 1 | LI-64-54A or<br>LI-64-66 | Suppression Chamber Water Level | Indicator -25" to +25" | | (1)(4) |
| 1 | NA | Control Rod Position | (2) | | (2)(4) |
| 2 | SRM A, B, C, D | Neutron Monitoring | Indicator and Recorder<br>0.1 to $10^{+6}$ cps<br>-100 to +10 sec. (period) | Downscale $\geq$ 3 cps<br>Retract permit $\geq$ 100 cps<br>Upscale HI $\leq 10^5$ cps<br>Upscale HI-HI$\leq 5 \times 10^5$cps<br>Period $\geq$ 30 sec. | (1)(3)(4) |
| 1 | LS-78-2A | Fuel Storage Pool level high | NA | $\leq$EL 663' 1/2" | (6)(7) |
| 1 | LS-78-2B | Fuel Storage Pool level low | NA | $\geq$EL 662' 7 1/2" | (6)(7) |
| 1 | TR-74-80 pT 17 | Fuel pool temperature | Recorder 0-600°F | $\leq$ 125°F | (6) (7) |

NOTES FOR TABLE 3.2.A

(1) If one of the instrument channels monitoring a parameter should become incapable of performing its intended function, all operations which could affect the associated system will be suspended until the item is returned to service.

(2) The control rod position indicator full-in switches will be operable for every control rod and will provide indication in the control room, or the control rod position will be verified to be full-in by visual observation. Control rod position indication to verify full-in position will be maintained in the control room.

(3) The following Source Range Monitoring Channels will be operable and will provide count rate indication and alarms in the control room:

SRM Channel A or C

SRM Channel B or D

The alarms in the control room will be as follows:

SRM TRIPS

| Trip Function | Trip Action |
|---|---|
| SRM upscale or inoperative | Annunciator, amber light |
| SRM Downscale | Annunciator, white light |

In addition, the SRM's have been placed in the non-coincidence scram mode to provide for protection against high neutron flux.

(4) Only required whenever irradiated fuel is in the reactor vessel.

(5) Only required whenever the reactor pressure vessel head is bolted in place.

(6) Only required when irradiated fuel is in the fuel pool.

(7) If this instrument channel is out of service, alternate manual means will be used until it is returned to service.

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION
(POST ACCIDENT INSTRUMENTATION)

| Instrument Channel | Calibration Frequency | Instrument Check |
|---|---|---|
| 1) Reactor Water Level | Once/6 months | Each Shift |
| 2) Reactor Pressure | Once/6 months | Each Shift |
| 3) Drywell Pressure | Once/6 months | Each Shift |
| 4) Drywell Temperature | Once/6 months | Each Shift |
| 5) Suppression Chamber Water Temperature | Once/6 months | Each Shift |
| 6) Suppression Chamber Water Level | Once/6 months | Once/Week |
| 7) Control Rod Position | NA | Each Shift (1) |
| 8) Neutron Monitoring (SRM) | Once/3 months (2) | Each Shift |
| 9) Drywell Pressure (PS-64-67) | Once/6 months | NA |
| 10) Fuel Pool Level (LS-78-2A, 2B) | Once/6 months | Once/Day |
| 11) Fuel Pool Tempenature (TR-74-80) | Once/6 months | Once/Day |

NOTES FOR TABLE 4.2.A

(1) Control rod position indication to verify the full-in position will be maintained in the control room or the rod position will be verified by visual observation.

This monitoring circuitry will be tested once a week.

(2) The SRM will be functionally tested on frequency of once per week.

3.2/4.2 BASES: PROTECTION INSTRUMENTATION (CONTINUED)

Control rod position indication to verify the full-in position will be maintained in the control room or the rod position will be verified by visual observation. Under the conditions to be maintained in Technical Specification 3.3, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions. A monitoring system will be provided in the main control room to alert the operator if any rod drift should occur.

The reactor water level will be nominally maintained at a level greater than 27" as indicated on LR-3-53. Above or below this level will give an alarm of "Reactor Water Level Abnormal." 27" indication on LR-3-53 corresponds to 555" above vessel zero.

In the unlikely event that loss of fuel pool level occurs, the low level alarm is set high enough above the fuel such that operator action can be taken to establish makeup before the level reaches 8-1/2 feet above the fuel. Refer to the specification 3.10 for fuel pool water conditions.

3.3 REACTIVITY CONTROL

DELETE PAGE 109

4.3 REACTIVITY CONTROL

3.3/4.3 BASES: REACTIVITY CONTROL

To prevent an inadvertent or spurious withdrawal of a control rod the directional control valves of each control rod have been electrically disarmed. As a further precaution, the valve in the drive water supply to each hydrauli control unit will be closed. In the unlikely event that a control rod does become withdrawn, two channels of the SRM's are required to be available for visual indication of neutron level. Although the SRM's may not immediately respond to a single rod movement, they would be adequate to monitor the approach to criticality. Also the SRM's are connected in the non-coincidence scram mode to provide rapid rod insertion from a high-high count rate on either SRM. Additionally, a manual scram will also be available. To assure that the control rods can be scrammed, the control rod accumulators are required to be charged with nitrogen and water pressure and a control rod drive pump is required to be in service. To provide additional indication of a control rod withdrawal, the control rod position indicator full-in switches will be functional for every control rod and will provide indication in the control room or the control rod

position may be verified to be full-in by visual observation.

While fuel is in the spent fuel pool

the control of reactivity is assured by the design of the spent fuel storage pool as described in FSAR, Chapter 10, Subsection 3.

LIMITING CONDITIONS FOR OPERATION

## .4 STANDBY LIQUID CONTROL SYSTEM

### Applicability

Applies to the operating status of the Standby Liquid Control System.

### Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

### Specification

A. Normal System Availability

1. The standby liquid control system shall be functional whenever fuel is in the reactor vessel.

SURVEILLANCE REQUIREMENTS

## 4.4 STANDBY LIQUID CONTROL SYSTEM

### Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

### Objective

To verify the operability of the Standby Liquid Control System.

### Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested.
2. At least once during each operating cycle:
    a. Check that the setting of the system relief valves is 1425 ± 75 psig.
    b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm

LIMITING CONDITIONS FOR OPERATION

.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the operational status of the core, suppression pool, and fuel pool cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. When irradiated fuel is in the reactor vessel, at least one core spray loop with one pump and available diesel generator shall be capable of delivering flow.

B. Residual Heat Removal System (RHRS) Containment and Shutdown Cooling

1. When irradiated fuel is in the reactor vessel at least two RHR loops with one pump per loop

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core, suppression pool, and fuel pool cooling systems when the corresponding limiting conditions for operation are in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing

| | Item | Frequency |
|---|---|---|
| a. | Pump operability | monthly |
| b. | Motor operated valve operability | monthly |

2. When it is determined that one core spray loop with one pump and available diesel generator is incapable of deliveri

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

shall be capable of manual operation. Diesel generators must also be available to power the pumps. A service water supply must be available.

C. Spent Fuel Pool Cooling

1. Whenever irradiated fuel is stored in the spent fuel pool, a cooling system for the spent fuel pool shall maintain the temperature of the fuel pool coolant $\leq$125°F.

2. When irradiated fuel is stored in the spent fuel pool, any combination of two pumps and associated heat exchangers from the spent fuel cooling or RHR supplemental cooling systems shall be available from different operable diesel generators to maintain fuel pool temperatures as speci-

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

flow, another core spray pump with an available diesel generator shall be selected, and all active components in the flow paths shall be immediately demonstrated to be capable of delivering flow.

B. Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

1. Residual Heat Removal System Testing

| | Item | Frequency |
|---|---|---|
| a. | Pump operability | monthly |
| b. | Motor operated valve operability | monthly |

2. When it is determined that one RHR pump (containment and suppression pool cooling) or associated heat exchanger is incapable of delivering flow and removing heat at a time when flow capability and heat removal are required, the remaining RHR pump and associated heat exchanger and available diesel generator, and all active components in the flow paths shall be demonstrated to be

fied in 3.5.C.1. When a fuel pool cooling pump is required to be operating or as a backup, the associated RBCCWS loop and service water system must be functional.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

D. RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

1. When a RHR pump is required to be operating or as a backup for supplemental cooling, an associated RHRSW pump must be functional and aligned to RHR header service corresponding to the selected RHR pump.

2. At all times, at least 2 RHRSW pumps shall be assigned to EECW header service with one pump assigned to each header.

Each pump shall be capable of automatic start in its normal D/G load sequencing mode of operation. Each pump shall be assigned to a separate diesel power supply.

3. Prior to restoration of any non-essential EECW loads that could result in exceeding the capacity of one RHRSW pump, a second RHRSW pump will be assigned to each EECW header.

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

capable of delivering flow and heat removal immediately and weekly thereafter until the inoperable RHR pump and associated heat exchanger is returned to service or an alternate pump and heat exchanger with an available diesel generator selected and verified.

C. Spent Fuel Pool Cooling

1. The spent fuel pool water temperature shall be checked and recorded at least every 8 hours.

2. When it is determined that the the RHR or fuel pool cooling pump for spent fuel pool cooling is incapable of heat removal, another RHR or fuel pool cooling pump capable of being supplied with diesel power shall be selected and all active components required for heat removal shall be demonstrated to be capable of delivering flow.

CORE, CONTAINMENT AND FUEL POOL COOLING SYSTEMS

Each pump must run continuously with its loss of voltage trips deactivated or it shall be capable of automatic start in its normal D/G load sequencing mode of operation. Each pump on the same header shall be assigned to a separate diesel power supply.

4. Two independent flow paths for water make up the spent fuel pool and the reactor vessel will be available from two RHRSW pumps capable of being supplied by separate diesel power.

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3. Routine surveillance for an operating or backup RHR or fuel pool cooling pump is as follows:

| Item | Frequency |
| --- | --- |
| a. Pump operability | monthly |
| b. Motor-operated valve operability | monthly |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|
| | D. RHR Service Water System (RHRSWS) And Emergency Equipment Cooling Water System (EECWS)<br><br>1. RHR Service Water System<br><br>Each of the required RHRSW pumps and associated essential control valves on the RHR heat exchanger headers shall be demonstrated to be functional once every month if not in continuous service. |

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

E. Maintenance of Filled Discharge Pipe

Whenever the core spray system or RHR systems are required to be functional, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The condensate or pressure suppression chamber head tank shall be aligned to serve the discharge piping of the RHR and CS pumps. The pressure indicators on the discharge piping of the RHR and CS pumps shall indicate not less than listed below.

| | |
|---|---|
| PI-75-20 | 48 psig |
| PI-75-48 | 48 psig |
| PI-74-51 | 48 psig |
| PI-74-65 | 48 psig |



4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

2. EECW System

Each of the RHRSW pumps assigned to EECW service and associated essential control valves on the EECW headers shall be demonstrated to function once every month.

3. When it is determined that one RHRSW pump and associated control valves on an RHR heat exchanger header are incapable of delivering flow at a time when flow delivery capability is required, another RHRSW pump and associated heat exchangers and available diesel generator and all active components in the flow paths shall be demonstrated to be capable of delivering flow immediately and weekly thereafter.

4. When it is determined that one RHRSW pump and associated

## 3.5.C Spent Fuel Pool Cooling (continued)

The decay heat removal requirements for a full core stored in the fuel pool can be conservatively met by the operation of one RHR pump and its associated RHR heat exchanger in the fuel pool cooling mode. The total heat load for this mode is estimated to be less than 20 percent of the heat exchanger capability under the required flow and temperature conditions.

## 3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

### 3.5.D RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump on one heat exchanger on each unit. One RHRSW pump is required for each unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allow ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump, if non-essential loads are valved out. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement two RHRSW pumps are assigned to EECW service and are aligned to separate supply headers. Testing activities require addition of any non-essential EECW loads which could exceed the capacity of one RHRSW pump, an additional pump will be assigned to service on each EECW header. Each of the required pumps will operate continuously (with loss of voltage trips deactivated) or they will be capable of automatic start in their normal diesel generator load sequencing mode of operation. The required RHRSW pumps are assigned to 4.16-kV shutdown boards which have associated operable diesel generators.

## 3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

### 3.5.E Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray and RHR system are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in a functional condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be nonfunctional for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow prior to any pump operation to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, optional head tanks located above the discharge line high point can supply makeup water for these systems. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and a minimum of 70 psig for a water level at the condensate head tank or 48 psig for the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

LIMITING CONDITIONS FOR OPERATIONS

3.7 CONTAINMENT SYSTEMS

B. Standy Gas Treatment

When the reactor zone ventilation system is removed from service, one train of the standby gas treatment system shall be in operation on the reactor building zone of the affected unit.

C. Reactor Building Ventilation

The reactor building zone for units 1 and 2 shall be ventilated by one supply and one exhaust fan per zone, except as specified in 3.7.B.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

B. Standby Gas Treatment

When required to be in service, operation of one train of the standby gas treatment system shall be verified and documented once per shift.

C. Reactor Building Ventilation

When required operation of the ventilation fans for the reactor building zone shall be verified daily.

## 3.7 BASES: CONTAINMENT SYSTEMS

### A. PRIMARY CONTAINMENT

This specification ensures indication of adequate information regarding status of the drywell pressure and temperature and suppression chamber water level and temperature when fuel is in the reactor. When fuel is removed this requirement is no longer necessary. Monitoring of information concerning these primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

### B. STANDBY GAS TREATMENT

Before making the normal reactor zone ventilation inoperable, one standby gas treatment train must be operating to provide a means to remove equipment heat and to maintain environmental temperature control in the affected reactor building zone.

### C. REACTOR BUILDING ZONES

Reactor building equipment heat removal and environmental temperature control will be provided by operation of the building ventilation systems. One ventilation supply fan and one ventilation exhaust fan in each reactor building zone will maintain ambient temperatures at an acceptable level.

3.10 CORE ALTERATIONS

2. Prior to fuel loading the SRM shall have an initial minimum count rate of 3 cps with all rods fully inserted

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2 feet above the top of the stored fuel.

2. Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be $\leq 125^{\circ}F$.

3. Fuel pool water shall be maintained within the following limits:

Conductivity, 10 umho/cm @ $25^{\circ}C$
& Chloride, 0.5 ppm

D. Reactor Building Crane

1. The reactor building crane shall be operable:

a. When a spent fuel cask is handled.



4.10 CORE ALTERATIONS

response.

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

2. Whenever irradiated fuel is stored in the spent fuel pool, the water temperature shall be recorded daily.

3. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

a. At least daily for conductivity and chloride ion content.

b. At least once per 8 hour shift for conductivity and chloride ion content when the fuel pool cleanup system is inoperable.

D. Reactor Building Crane

1. The following operational checks and inspections shall be performed on the reactor building crane prior to

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability

Applies to the loading of fuel in the reactor pressure vessel.

Objective

To prevent criticality during the loading of fuel in the reactor pressure vessel.

Specification

A. Reactivity Control

1. The reactor shall be kept in cold shutdown and all control rods fully inserted during fuel loading while more than one fuel assembly is in the reactor.

2. The reactor mode switch shall be locked in the "SHUTDOWN" position during fuel loading.

B. Core Monitoring

During fuel loading two channels of the SRM's, each on separate power supplies, shall be operable. For a SRM to be considered operable, the following conditions must be satisfied:

1. The SRM shall be fully inserted into the core.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic verification of rod position and testing of instrumentation used during fuel loading.

Objective

To verify full insertion of control rods and operability of instrumentation during fuel loading.

Specification

A. Reactivity Control

1. Surveillance to verify full insertion of all control rods is specified in 4.3.

2. Prior to loading fuel, and daily thereafter, verify that the reactor mode switch is locked in the "SHUTDOWN" position.

B. Core Monitoring

Prior to the loading of fuel in the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for

3.10/4.10 BASES: CORE ALTERATIONS

B. Core Monitoring

The SRM's are provided to monitor the core during shutdown and to monitor core reactivity and guide the operator while fuel is being loaded in the reactor pressure vessel.

Requiring the SRM's to be functionally tested prior to fuel loading assures that the SRM's will be operable at the start of fuel loading. The daily response check of the SRM's ensures their continued operability.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

## 3.11 FIRE PROTECTION SYSTEMS

Applicability:

Applies to operating status of the high pressure water and $CO_2$ fire protection systems for the reactor building, diesel generator buildings, control bay, intake pumping station, cable tunnel to the intake pumping station, and the fixed spray system for cable trays along the south wall of the turbine buildng, elevation 586.

Objective:

To assure availability of Fire Protection Systems.

Specification:

A. High Pressure Fire Protection System

1. The High Pressure Fire Protection System shall have:

   a. Two (2) high pressure fire pumps operable and aligned to the high pressure fire header.

   b. Automatic initiation logic operable.

## 4.11 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the surveillance requirements of the high pressure water and $CO_2$ fire protection systems for the reactor building, diesel generator buildings, control bay, intake pumping station, cable tunnel to the intake pumping station, and the fixed spray system for cable trays along the south wall of the turbine building, elevation 586 when the corresponding limiting conditions for operation are in effect.

Objective:

To verify the operability of the Fire Protection Systems.

Specification:

A. High Pressure Fire Protection System

1. High Pressure Fire Protection System Testing:

| Item | Frequency |
|---|---|
| a. Simulated automatic and manual actuation of high pressure pumps | Once/year |
| b. Pump Operability | Once/month |
| c. Automatic valve operability | Once/3 months |
| d. Pump capability | Once/3 years |

## 3.11 FIRE PROTECTION SYSTEMS

## 4.11 FIRE PROTECTION SYSTEMS

| | | |
|---|---|---|
| | checked to be 2664 gpm at 250 feet head | |
| e. | Spray header and nozzle inspection for blockage | Once/year |
| f. | System flush in conjunction with semi-annual addition of biocide to the Raw Cooling Water System | Twice/year |
| g. | Building hydraulic performance verification | Once/3 years |
| h. | Yard loop and cooling tower loop hydraulic performance verification | Once/year |

## 3.11 FIRE PROTECTION SYSTEMS

2. If specification 3.11.A.1.a or 3.11.A.1.b cannot be met, a patrolling fire watch with portable fire equipment available shall be established to insure that each area where protection is lost is checked hourly.

3. If only one high pressure fire pump is operable, the reactors may remain in operation for a period not to exceed 7 days, provided the requirements of specification 3.11.A.1.b above are met.

4. If specification 3.11.A.3 cannot be met, the reactors shall be placed in the cold shutdown condition in 24 hours.

5. Removal of any component in the High Pressure Fire System from service for any reason other than testing or emergency operations shall require Plant Superintendent approval.

6. The Raw Service Water storage tank level shall be maintained above level 723'7" by the raw service water pumps.

## 4.11 FIRE PROTECTION SYSTEMS

2. When it is determined that only one pump is operable, that pump shall be demonstrated operable immediately and daily thereafter until specification 3.11.A.1.a can be met.

3. Raw Service Water System Testing

| Item | Frequency |
|---|---|
| Simulated automatic and manual actuation of raw service water pumps and operation of tank level switches | Once/year |

4. The high pressure fire protection system pressure shall be logged daily.

5. Principal header and component isolation valves shall be checked open at intervals no greater than three months.

## 3.11 FIRE PROTECTION SYSTEMS

7. If specification 3.11.A.6 cannot be met a fire pump shall be started and run continuously until the raw service water pumps can maintain a raw service water storage tank level above 723'7".

8. The fire protection water distribution system shall have a minimum capacity of 2664 gpm at 250' head.

9. The fire protection system shall be capable of supplying the individual loads listed in Table 3.11.A.

## 4.11 FIRE PROTECTION SYSTEMS

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

## 3.11 FIRE PROTECTION SYSTEMS

B. $CO_2$ Fire Protection System

1. The $CO_2$ Fire Protection System shall be operable:

   a. With a minimum of 8-1/2 tons (0.5 Tank) $CO_2$ in storage units 1 and 2.

   b. With a minimum of 3 tons (0.5 Tank) $CO_2$ storage unit 3.

   c. Automatic initiation logic operable.

2. If specification 3.11.B.1.a or 3.11.B.1.b or 3.11.B.1.c cannot be met, a patrolling fire watch with portable fire equipment shall be established to ensure that each area where protection is lost is checked hourly.

3. If specifications 3.11.B.1.a, 3.11.B.1.b, or 3.11.B.1.c are not met within 7 days, the affected unit(s) shall be in cold shutdown within 24 hours.

## 4.11 FIRE PROTECTION SYSTEMS

B. $CO_2$ Fire Protection System

1. $CO_2$ Fire Protection Testing:

| Item | Frequency |
|---|---|
| a. Simulated automatic and manual actuation | Once/year |
| b. Storage tank pressure and level | Checked daily |
| d. $CO_2$ Spray header and nozzle inspection for blockage | Once/3 years |

2. When the cable spreading room $CO_2$ Fire Protection is inoperable, one 125-pound (or larger) portable fire extinguisher shall be placed at each entrance.

## 3.11 FIRE PROTECTION SYSTEMS

4. If $CO_2$ fire protection is lost to a cable spreading room or to any diesel generator building area a continuous fire watch shall be established immediately and shall be continued until $CO_2$ fire protection is restored.

5. Removal of any component in the $CO_2$ Fire Protection System from service for any reason other than testing or emergency operations shall require Plant Superintendent approval.

C. Fire Detectors

1. The fire detection system's heat and smoke detectors for all protected zones shall be operable except that one detector for a given protected zone may be inoperable for a period no greater than 30 days.

2. If specification 3.11.C.1 cannot be met, a patrolling fire watch will be established to ensure that each protected zone or area with inoperable detectors is checked at intervals no greater than one each hour.

## 4.11 FIRE PROTECTION SYSTEMS

C. Fire Detectors

1. All heat and smoke detectors shall be tested in accordance with industrial standards or other approved methods semiannually.

2. The non-Class A supervised detector circuitry for those detectors which provide alarm only will be tested once each month by actuating the detector at the end of the line or end of the branch such that the largest number of circuit conductors will be checked.

## 3.11 FIRE PROTECTION SYSTEMS

D. A roving fire watch will tour each area in which automatic fire suppression systems are to be installed (as described in the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2," Section X) at intervals no greater than 2 hours. A keyclock recording type system shall be used to monitor the routes of the roving fire watch. The patrol will be discontinued as the automatic suppression systems are installed and made operable for each specified area.

## 4.11 FIRE PROTECTION SYSTEMS

3. The class A supervised detector alarm circuits will be tested once each two months at the local panels.

4. The circuits between the local panels in 4.11.C.3 and the main control room will be tested monthly.

5. Smoke detector sensitivity will be checked in accordance with manufacturer's instruction annually.

D. A monthly walk-through by the Safety Engineer will be made to visually inspect the plant fire protection system for signs of damage, deterioration, or abnormal conditions which could jeopardize proper operation of the system.

## 3.11 FIRE PROTECTION SYSTEMS

E. Fire Protection System Inspection

1. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified TVA personnel or an outside fire protection firm.

2. An inspection and audit by an outside qualified fire consultant will be performed at intervals no greater than 3 years. (The first inspection and audit will be during the period of May-June 1979.)

F. If it becomes necessary to breach a fire stop, an attendant shall be posted on each aide of the open penetration until work is completed and the penetration is resealed.

G. The minimum in-plant fire protection organization and duties shall be as depicted in Figure 6.3-1.

## 4.11 FIRE PROTECTION SYSTEMS

E. Fire Protection Systems Inspection

Any inspection or audit will review and evaluate the effectiveness of fire prevention and protection by physical inspection of plant facilities, systems, and equipment as related to fire safety. Evaluations will be made of, but not necessarily limited to, the following:

Administrative control documentation, maintenance of fire related records, physical plant inspection, related historical research and application, and management interviews.

## 3.11 FIRE PROTECTION SYSTEMS

H. A minimum of fifteen air masks and thirty 500 cubic inch air cylinders shall be available at all times except that a time period of 48 hours following emergency use is allowed to permit recharging or replacing.

I. A continuous fire watch shall be stationed in the immediate vicinity where work involving open flame welding, or burning is in progress.

J. There shall be no use of open flame, welding, or burning in the cable spreading room unless the reactor is in the cold shutdown condition.

## 4.11 FIRE PROTECTION SYSTEMS

TABLE 3.11.A

FIRE PROTECTION SYSTEM HYDRAULIC REQUIREMENTS

| Station | Flow Required (gpm) | Residual Pressure (psig) |
|---|---|---|
| 1. Reactor Building Roof | | |
| A. Valve 26-849 | 200 | 65 |
| B. Valve 26-889 | 200 | 65 |
| 2. Refuel Floor | | |
| A. Valve 26-835 | 75 | 70 |
| B. Valve 26-843 | 75 | 70 |
| C. Valve 26-870 | 75 | 70 |
| D. Valve 26-865 | 75 | 70 |
| E. Valve 26-876 | 75 | 70 |
| F. Valve 26-888 | 75 | 70 |
| G. Valve 26-898 | 75 | 70 |
| 3. Cable Tray Fixed Spray | | |
| A. Unit 1 - Station I | 300 | 70 |
| B. Unit 1 - Station II | 200 | 70 |
| C. Unit 1 - Station III | 180 | 65 |
| D. Unit 2 - Station II | 200 | 70 |
| E. Unit 2 - Station III | 200 | 70 |
| F. Unit 3 - Station II | 200 | 70 |
| G. Unit 3 - Station III | 265 | 75 |
| H. Turbine Building | 30 | 55 |
| 4. Diesel Generator Buildings | | |
| A. Valve 26-1032 | 75 | 70 |
| B. Valve 26-1069 | 75 | 70 |
| 5. Pump Intake Station | | |
| A. Valve 26-1076 | 75 | 70 |

TABLE 3.11.A

FIRE PROTECTION SYSTEM HYDRAULIC REQUIREMENTS

| Station | Flow Required (gpm) | Residual Pressure (psig) |
|---|---|---|
| 6. Control Bay | | |
| A. Valve 26-1076 | 75 | 70 |
| 7. Yard Loop (1) | | |
| A. Hydrant at valve 0-26-526 | 500 | 65 |
| B. Hydrant at valve 0-26-530 | 500 | 65 |
| 8. Cooling Tower Loop | | |
| A. Hydrant at valve 0-26-1023-6 | 500 | 65 |

260k

Note (1) Yard hydrants and the cooling tower hydrant are to be tested using the longest

## 3.11 BASES

The High Pressure Fire and $CO_2$ Fire Protection specifications are provided in order to meet the preestablished levels of operability during a fire in either or all of the three units. Requiring a patrolling fire watch with portable fire equipment if the automatic initiation is lost will provide (as does the automatic system) for early reporting and immediate fire fighting capability in the event of a fire occurrence.

The High pressure Fire Protection System is supplied by three pumps aligned to the high pressure fire header. The reactors may remain in operation for a period not to exceed 7 days if two pumps are out of service. It at least two pumps are not made operable in seven days or if all pumps are lost during this seven day period, the reactors will be placed in the cold shutdown condition within 24 hours.

For the areas of applicability, the fire protection water distribution system minimum capacity of 2664 gpm at 250' head at the fire pump discharge consists of the following design loads:

| | | |
|---|---|---|
| 1. | Sprinkler System (0.30 gpm/ft²/4440 ft² area) | 1332 gpm |
| 2. | 1 1/2" Hand Hose Lines | 200 gpm |
| 3. | Raw Service Water Load | 1132 gpm |
| | TOTAL | 2664 gpm |

The $CO_2$ Fire Protection System is considered operable with a minimum of 8 1/2 tons (0.5 tank) $CO_2$ in storage for units 1 and 2; and a minimum of 3 tons (0.5 tank) $CO_2$ in storage for unit 3. An immediate and continuous fire watch in the cable spreading room or any diesel generator building area will be established if $CO_2$ fire protection is lost in this room and will continue until $CO_2$ fire protection is restored.

To assure close supervision of fire protection system activities, the removal from service of any component in either the High Pressure Fire System or the $CO_2$ Fire Protection System for any reason other than testing or emergency operations will require Plant Superintendent approval.

Early reporting and immediate fire fighting capability in the event of a fire occurrence will be provided (as with the automatic system) by requiring a patrolling fire watch if more than one detector for a given protected zone is inoperable.

A roving fire watch for areas in which automatic fire suppression systems are to be installed will provide additional interim fire protection for areas that have been determined to need additional protection.

The fire protection system is designed to supply the required flow and pressure to an individual load listed on Table 3.11.A while maintaining a design raw service water load of 1132 gpm.

## 4.11 BASES

Periodic testing of both the High Pressure Fire System and the $CO_2$ Fire Protection System will provide positive indication of their operability. If only one of the pumps supplying the High Pressure Fire System is operable, the pump that is operable will be checked immediately and daily thereafter to demonstrate operability. If the $CO_2$ Fire Protection System becomes inoperable in the cable spreading room, one 125-pound (or larger) fire extinguishere will be placed at each entrance to the cable spreading room.

Wet fire header flushing, spray header inspection for blockage, and nozzle inspection for blockage will prevent, detect, and remove buildup of sludge or other material to ensure continued operability. System flushes in conjunction with the semiannual addition of biocide to the Raw Cooling Water System will help prevent the growth of crustaceans which could reduce nozzle discharge.

Semiannual tests of heat and smoke detectors are in accordance with the NFPA code.

With the exception of continuous strip heat detectors panels, all non-class A supervised detector circuits which provide alarm only are hardwired through conduits and/or cable trays from the detector to the main control room alarm panels with no active components between. Non-class A circuits also actuate the HPCI water-fog system, the $CO_2$ system in the diesel generator buildings, and isolate ventilation in shutdown board rooms. The test frequency and methods specified are justified for the following reasons:

1. An analysis was made of worst-case fire detection circuits at Browns Ferry to determine the probability of no undetected failure of the circuits occurring between system test times as specified in the surveillance requirements. A circuit is defined as the wire connections and components that affect transmission of an alarm signal between the fire detectors and the control room annunciator. Three circuits were analyzed which were representative of an alarm-only circuit, a water-fog circuit, and a $CO_2$ circuit. The spreading room B smoke detector was selected as the worst-case alarm-only circuit because it had the largest number of wires and connections in a single circuit. The HPCI water-fog circuit was selected for analysis because it is the only water-fog circuit in the area of applicability for technical specifications. The Standby Diesel Generator Room A $CO_2$

circuit was selected because it contained 2 out of 3 detector logic, the most complicated $CO_2$ circuit logic. Calculations were based on failure rates for wires, connections, and circuit components as shown in Appendix III of WASH-1400. Failure rates were considered for the following circuit components:

1. Open circuit
2. Short to ground
3. Short to power
4. Timing motor failure to start
5. Relay failure to energize
6. Normally open contact failure to close
7. Normally open or normally closed contact short
8. Normally closed contact opening
9. Timing switch failure to transfer

The calculated probabilities (Pf) for no undetected failure of the circuits occurring were as follows, based on the specified test frequency.

| AREA | TEST FREQUENCY | Pf |
|---|---|---|
| Spreading Room B | One Month | 0.975287 |
| HPCI Water Fog | Six Months | 0.977175 |
| Standby Diesel Gen Room A $CO_2$ | Six Months | 0.957595 |

The worst case of the three areas considered is Spreading Room B. The probability of undected failure is approximately 1/40, which means that one undetected failure will occur on the average every 40 months over an extended period of time and that the failure could exist up to one month. The frequency of testing is thus much greater than the frequency of failure and produces circuits with adequate reliability.

2. Circuits checks by initiation of end of the line or end of the branch detectors will more thoroughly test the parallel curcuits than testing on a rotating detector basis. This test is not a detector test, but is a test to simulate the effect of electrical supervision as defined in the NFPA code.*

3. Testing of circuits which actuate $CO_2$ , water, or ventilation systems requires disabling the automatic feature of the fire protection system for the area. A surveillance program which disabled these circuits monthly would significantly reduce the ability of these circuits to provide fire suppression.

*Ref: NFPA Code 72D-9, paragraph 1111, Code 72D-15, paragraph 1312 for definition of Class A systems, and Code 72A-18, Article 240.

4. Daily tests of annunciation lights and audible devices are performed as a routine operation function.

5. The $CO_2$ system manufacturer recommends semiannual testing of $CO_2$ system fire detection circuits.

Figure 6.3-1 describes the in-plant fire protection organization including the roving fire watch. In addition, other operating personnel periodically inspect the plant during their normal operating activities for fire hazards and other abnormal conditions.

Smoke detectors will be tested "in-place" using inert freon gas applied by a pyrotronics type applicator which is accepted throughout the industrial fire protection industry for testing products of combustion detectors or by use of the MSA chemical smoke generators. At the present time the manufacturers have only approved the use of "punk" for creating smoke. TVA will not use "punk" for testing smoke detectors.

4. Emergency conditions involving potential or actual release of radioactivity.

5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

6. Surveillance and testing requirements.

7. Radiation control procedures.

8. Radiological Emergency Plan implementing procedures.

9. Plant security program implementing procedures.

10. Fire protection and prevention procedures.

B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation. Temporary changes to a procedure which do not change the intent of the approved procedure may be made by a member of the plant staff knowledgeable in the area affected by the procedure except that temporary changes to those items listed above except item 5 require the additional approval of a member of the plant staff who holds a Senior Reactor Operator license on the unit affected. Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent.

C. Drills on actions to be taken under emergency conditions involving release of radioactivity are specified in the radiological emergency plan and shall be conducted annually. Annual drills shall also be conducted on the actions to be taken following failures of safety related systems or components.

D. Radiation Control Procedures

Radiation Control Procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20 except in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

1. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled by issuance of a special work permit. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. DPR-63

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY UNIT NO. 2

DOCKET NO. 50-260

FUEL LOADING

## Introduction

On February 23, 1976, the Nuclear Regulatory Commission (NRC) issued its "Safety Evaluation by the Division of Operating Reactors Supporting the Operation After the Restoration and Modification of the Browns Ferry Nuclear Plant, Units 1 and 2 Following the March 22, 1975 Fire". This safety evaluation addressed acceptability of the modifications being performed at Browns Ferry to provide adequate protection against both divisions of engineered safety features from being rendered inoperable by a fire, thereby ensuring the safe shutdown capability of the plant. There were some open items identified in that safety evaluation and, as a result of the review of the Advisory Committee on Reactor Safeguards (ACRS), there were some items that required further response or evaluation. These are addressed in Supplement No. 1 to the February 23rd Safety Evaluation which was issued June 18, 1976.

On May 28, 1976, TVA made application to load fuel in Units 1 and 2 and proposed changes to the Interim Technical Specifications to cover that activity. On June 1, 1976, TVA submitted revisions to those proposed changes. By order dated May 21, 1976, the Atomic Safety and Licensing Board (ASLB) authorized the Director of Nuclear Reactor Regulation to authorize fuel loading of Units 1 and 2 subject to certain conditions specified therein. These conditions included: (1) that the control rod drives would be disabled with the control rods in the fully inserted position and (2) that the valves of all systems which could allow loss of water from the primary system would be disabled in a position that prevented such loss.

These conditions resulted from the fact that issues raised by certain contentions before the ASLB have not yet been adjudicated. These contentions relate to whether the modification and restoration are sufficient or have been properly performed to assure adequate protection against severe accidents resulting from insufficient or improperly performed work. As pointed out in the affidavit supplied by the Staff with the "NRC Staff Response to Licensee's Motion for an Order Authorizing Fuel Loading and Operation" dated May 5, 1976, the Browns Ferry Nuclear Plant Units 1 and 2 may be refueled without risk of serious accident, even if no reliance were to be placed on the proper operation of systems whose cabling and circuitry have been restored or whose design has been modified after the March 22, 1975 fire. The ASLB Order of May 21, 1976 accepted the staff position and authorized the Director of Nuclear Reactor Regulation to make appropriate findings and to authorize fuel loading in accordance with conditions proposed by the staff.

Evaluation

The restoration and modifications of Browns Ferry Unit 2 have been completed in accordance with the requirements of TVA's "Plant for Evaluation Repair and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975 and revisions thereto up to and including Revision 45. These activities have been evaluated in the NRC's Safety Evaluation dated February 23, 1976 and certain additional matters relating to fire protection required are evaluated in the supplement dated June 18, 1976. The adequacy of the restoration work and the ability of systems that were damaged or potentially damaged by the fire or which have been modified has been demonstrated by the satisfactory completion of the preoperational retest program. With plant systems restored to original design conditions and functional requirements and with modified cabling and circuitry as discussed in the staff SER's dated September 2, 1975 complted, the evaluations carried out by the staff to support the original operating license are applicable to demonstrate that there is reasonable assurance of no undue risk to public health and safety in connection with the operation of Unit 2, as discussed in those SER's. In addition, after the fire, the staff has directed substantial additional attention to enhanced fire protection capability of the plant. Our evaluation of fire protection provided in connection with the restoration of the facility is set forth in the SER dated February 23, 1976 and its Supplement dated June 18, 1976. The remaining fire protection evaluation and work activities to be completed before return to substantial power operation, are not required to assure adequate protection during fuel loading.

The fuel is presently in the spent fuel storage pool. At the outset of the refueling operation, the reactor cavity and transfer canal are filled with water to levels of approximately 60 feet above the top of the fuel assemblies when they are in the reactor vessel core. The fuel assemblies are removed from the storage racks and are transferred under water through the refueling transfer canal to the reactor and are inserted into the core support assembly. This transfer takes place one at a time using the fuel grapple. Each element has been marked for identification and will be returned to the position in the core which it occupied during operation. The systems and equipment used in connection with movement from the fuel pool to the reactor core have been tested to assure that they function properly.

Inadvertent excessive withdrawal of a fuel element above the refueling water level is prevented by the fuel loading crane interlocks and by the mechanical design of the fuel handling equipment. The interlocks cut off power to the hoist while there is still in excess of eight feet of water above the fuel. This interlock which prevents excessive withdrawal was not damaged by the fire and has been retested to assure that it functions properly.

With the fuel assemblies moved one at a time into a core position with control rods fully inserted, the accidental dropping of fuel element could not result in criticality, but it may damage the cladding or result in release of fission products contained in the fuel element. However, since the element is under not less than 8 feet of water at all times and would be under at least 27 feet of water when it struck the bottom of the spent fuel pool, the release of fission products from the refueling water to the containment atmosphere would be very small. The various safety systems to reduce the release of radioactivity from the reactor to the environment have been restored and those that were potentially affected by the fire have been functionally tested to assure that they will properly perform. However, even without credit for any active safety systems, the potential two-hour site boundary doses are well below 0.1 rem.

The basic Final Safety Analysis Report for the facility upon which the staff Safety Evaluation Reports supporting the issuance of the operating license on June 28, 1974, were based, discusses the radiological protection program at the facility and demonstrates that adequate protection has been provided against occupational exposure during the refueling operation.

After the fuel has been placed in the reactor core it will be cooled by recirculation of water through the RHR system. Redundant RHR pumps and heat exchanges are available and all components including their power and control systems and circuitry have been restored and tested to function properly. Although certain fire protection evaluation has not yet been completed as discussed in the SER Supplement dated June 18, 1976, the modifications to the power circuitry of the shutdown bus and alternate feeds to the 480V diesel auxiliary boards have substantially enhanced the reliability of the RHR systems.

The shutdown bus being moved was installed in conduit and was routed over the other division's tray system consequently suffering damage in the fire. This shutdown bus has been relocated and that it is no longer in a zone of influence of any opposit division cable trays. Consequently, it is no longer physically susceptible to a fire similar to the March 22, 1975 fire which affected both sets of power supply to certain shutdown boards. Since the shutdown boards have transfer controls which enable power and control to be supplied to the RHR system locally rather than protection evaluation and work activities in this area has little or no effect on assurance of continued functioning of the RHR system.

Other changes were made in order to provide individual normal feeders to the 4KV/480V transformers. One change will eliminate sharing, between units, of the 4KV feeder that was the normal supply to 480V shutdown borad 1B through transformer TS1B and the alternate supply to 480V shutdown boards 2A and 2B through TS2E. The alternate 4KV diesel auxiliary borads A and B were relocated to afford divisional separation.

In addition, even if the substantial amount of water above the vessel in the reactor cavity is disregarded; the quantity of water above the fuel in the vessel is in excess of 8000 cubic feet. The present product decay heat inventory in the core from Browns Ferry Unit 1 is less than 0.5 Mw thermal or 1.706 x $10^6$BTU/Hr. The decay heat inventory in the core for Unit 2 is even smaller. Consequently, even if one assumed that the redundant RHR systems all failed to provide cooling to the core in the vessel and the water level was at the vessel flange (25 feet below the refueling level, it would take in excess of 300 hours before evaporation and boiloff would reach the top of the fuel. Furthermore, at such a low decay heat level only about 3 GPM is needed to supply sufficient cooling water for boiloff in order to keep the fuel covered. This amount can be provided through an ordinary garden hose. At Browns Ferry a fire hose at the operating floor above the reactor cavity will supply at least 100 gpm.

In order to assume that refueling operation could be carried out safely without entailing the risk of a severe accident resulting from insufficient or improperly performed modifications or restoration work, since that matter was still in controvery before the ASLB, the staff proposed two conditions to eliminate reliance on systems which had been restored or modified after the fire, to prevent severe accidents.

The ASLB has authorized the Director of Nuclear Reactor Regulation to authorize fuel loading in accordance with staff's proposed conditions. Since these two conditions are somewhat unusual for fuel loading, their effect on refueling safety is discussed below.

The first condition involves disabling the control rod drive mechanisms in the fully inserted position. This prevents the use of the normal fuel loading procedure of checking the shutdown margin after completion of the loading of each 4 assembly cell surrounding a control rod. However, this is not a normal fuel loading in that all the fuel assemblies are being replaced in the same core position that they occupied during the operation of the core prior to the March 22, 1975 fire. This core has been measured for criticality and shutdown margins during the course of operation in its first cycle prior to the fire. These measurements demonstrated that the required shutdown margin is available. Administrative, redundant checks throughout the fuel loading process provide assurance that the fuel assemblies are replaced in the proper positions. The fuel assemblies have identification symbols that are double checked and recorded and the proper placement in the core is observed and verified correct by at least two members of the loading crew. Four source neutron monitoring channels detect the neutron flux level produced by the subcritical multiplication in the fuel throughout the fuel loading steps. From the observation of the stable count rates on these channels, the operator can determine that the core is subcritical and by drawing plots of the inverse source multiplication curves he gets an indication that loading an additional number of assemblies will not produce criticality. The loading of each assembly proceeds in a controlled deliberate manner so that observation of the count rates on the source channels can be used to stop any insertion. Even if multiple fuel loading errors were made, inadvertent criticality would not occur since the infinite multiplication factor for each of the three types of fuel in this core are less than unity at all exposures with a controlled core, i.e., all rods fully inserted. Therefore, an array consisting of any combination of these three types of fuel would also be subcritical in the fully rodded condition. Based on the foregoing evaluation, we have concluded that fuel loading can proceed safely with the condition that all rods remain fully inserted.

The second condition involves ensuring that systems connected to the primary system cannot drain water from the primary system by disabling certain valves in a position that prevents such draining from occurring. This condition is identical to that implemented by the May 9, 1975 and June 13, 1975 amendments that were issued prior to unloading the fuel from the core. The safety evaluations that accompanied those amendments provide the basis for the acceptability of these precautions. This involves the fact that, in the cold, shutdown condition, the operation of these valves to a position other than that required for normal decay heat cooling is not required. Therefore, the valves indicated in the May 1, 1975 safety analysis (Part VI, Section E of the "Plan" Appendix A) may be disabled in the proper position for normal decay heat cooling without any adverse effects and by so doing assurance is gained that misoperation or spurious operation will not lead to loss of primary coolant.

Conclusion

Based on the foregoing evaluation, we conclude that Unit 2 of the Browns Ferry Plant has had all systems and components needed for fuel loading, that were adversely affected or potentially adversely affected by the March 22, 1975 fire, restored and verified by functional testing to their original design condition; that the remaining fire protection evaluation and modification do not affect safety of fuel loading; and that implementing of the special conditions imposed by ASLB order dated May 21, 1976 during fuel loading will not cause a reduction in any margin of safety nor increase the consequences of any previously analyzed accident nor introduce any new type of accident not previously evaluated. Therefore, we conclude that fuel loading of Unit 2 can proceed with reasonable assurance that the health and safety of the public will not be endangered.

Date: June 18, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-52, issued to Tennessee Valley Authority which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Unit No. 2, located in Limestone County, Alabama. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to allow loading the fuel of Unit No. 2 in the Unit No. 2 reactor vessel. Operation is not authorized by this amendment.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR § 51.5(d)(4) an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connections with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendment dated May 28, 1976 as supplemented June 1, 1976, (2) Amendment No. 20 to License No. DPR-52, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. 20555 and at the Athens Public Library, South Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of June, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors