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

OCT 29 1990

Docket No. 50-328

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

Dear Mr. Kingsley:

SUBJECT: UPPER HEAD INJECTION SYSTEM REMOVAL, BORON INJECTION TANK DEACTIVATION, AND CLOSEOUT OF TEMPORARY 50.46(a)(1) EXEMPTION (TAC 75748, 75750, 76678) (TS 89-25 and 89-26) - SEQUOYAH NUCLEAR PLANT, UNIT 2

The Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. This amendment is in response to your two applications dated January 12, 1990.

This amendment revises the Sequoyah Nuclear Plant, Unit 2, Technical Specifications (TSs) to account for the removal of the upper head injection system (UHIS) and the deactivation of the boron injection tank (BIT) at Unit 2 during the current Unit 2 Cycle 4 refueling outage. The changes to the TSs for UHIS removal delete TS 3/4.5.1.2 on the UHIS; revise Tables 3.4-1, 3.6-1, and 3.6-2 on the UHIS reactor coolant pressure isolation valves, penetrations, and containment isolation valves; revise the operability requirements in TS 3/4.5.1.1 on the cold leg injection accumulators; decrease the minimum flow rates in Surveillance Requirement (SR) 4.5.2.h for emergency core cooling systems; and revise TS 3/4.2.2 to reduce the peaking factor limit in the heat flux hot channel factor.

For the BIT deactivation, the changes to the TSs increase the boron concentration in the refueling water storage tank in TSs 3/4.1.2.5, 3/4.1.2.6 and 3/4.5.5; increase the boron concentration in the cold leg injection accumulator in TS 3/4.5.1.1; delete TS 3/4.5.4 on the BIT system; and increase the minimum volume of the boric acid storage system in TS 3/4.1.2.6. In addition, the reference to boron injection throttle valves will be changed to charging pump throttle valves in SR 4.5.2.g.

There are also changes to the Bases of the affected TSs and the index of the TSs. The changes for Sequoyah Unit 1 were issued by the staff's letter dated May 11, 1990 during the Unit 1 Cycle 4 refueling outage.

The changes to TS 3/4.2.2 account for the analysis required by 10 CFR 50.46. As discussed in your letter dated March 5, 1990, the analysis submitted in the application for UHIS removal (TS 89-25) fulfills the requirements in the exemption to 10 CFR 50.46 issued by the staff on January 26, 1989. Therefore, the requirement in the exemption that no more than five percent of the steam generator tubes can be plugged no longer applies to the operation of Unit 2. This limitation was removed from the operation of Unit 1 by the staff's letter dated May 11, 1990.



DFol
1/1

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

Sincerely,

Original signed by:

Jack N. Donohew, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 131 to
   License No. DPR-79
2. Safety Evaluation

cc w/enclosures:
See next page

| OFC | :PDII-4 | :PDII-4 | :OGC | :PDII-4/ADD | :PDII-4/D | : | : |
|---|---|---|---|---|---|---|---|
| NAME | :MKrebs MK | :JDonohew | : PAJ | :SBlack | :FHebdon | : | : |
| DATE | :10/10/90 | :10/11/90 | :10/11/90 | 10/29/90 | :10/29/90 | : | : |

OFFICIAL RECORD COPY
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. DPR-79

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

   A. The two applications for amendment by Tennessee Valley Authority (the licensee) dated January 12, 1990, comply 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-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne Black/for

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance:
October 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

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 marginal lines indicating the area of change. Overleaf pages* are provided to maintain document completeness.

| REMOVE | INSERT |
|---|---|
| VII | VII |
| VIII | VIII* |
| 3/4 1-11 | 3/4 1-11 |
| 3/4 1-12 | 3/4 1-12 |
| 3/4 2-4 | 3/4 2-4 |
| 3/4 2-5 | 3/4 2-5 |
| 3/4 2-6 | 3/4 2-6 |
| 3/4 2-6a | 3/4 2-6a |
| 3/4 2-7 | 3/4 2-7 |
| 3/4 4-20 | 3/4 4-20 |
| 3/4 5-1 | 3/4 5-1 |
| 3/4 5-3 | 3/4 5-3 |
| 3/4 5-4 | - - - |
| 3/4 5-5 | 3/4 5-4 |
| 3/4 5-6 | 3/4 5-5 |
| 3/4 5-7 | 3/4 5-6 |
| 3/4 5-8 | 3/4 5-7 |
| 3/4 5-9 | 3/4 5-8 |
| 3/4 5-10 | 3/4 5-9 |
| 3/4 5-11 | 3/4 5-10 |
| 3/4 5-12 | - - - |
| 3/4 5-13 | 3/4 5-11 |
| 3/4 6-6a | 3/4 6-6a |
| 3/4 6-21 | 3/4 6-21 |
| B 3/4 1-3 | B 3/4 1-3 |
| B 3/4 2-1 | B 3/4 2-1 |
| B 3/4 5-1 | B 3/4 5-1 |
| B 3/4 5-2 | B 3/4 5-2 |
| B 3/4 5-3 | --- |

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## BORATED WATER SOURCE - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and at least one associated heat tracing system with:

   1. A minimum contained borated water volume of 2175 gallons,
   2. Between 20,000 and 22,500 ppm of boron, and
   3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 35,443 gallons,
   2. A minimum boron concentration of 2500 ppm, and
   3. A minimum solution temperature of 60°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

### SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration of the water,
   2. Verifying the contained borated water volume, and
   3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water.

## BORATED WATER SOURCES - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A boric acid storage system and at least one associated heat tracing system with:

1. A minimum contained borated water volume of 7176 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A contained borated water volume of between 370,000 and 375,000 gallons,
2. Between 2500 and 2700 ppm of boron, and
3. A minimum solution temperature of 60°F.
4. A maximum solution temperature of 105°F.

APPLICABILITY: Modes 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

## 3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

### LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{2.32}{0.5}\right] [K(Z)] \text{ for } P \leq 0.5$$

$$\text{where } P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta T Trip Setpoints (value of $K_4$) have been reduced at least 1% (in $\Delta T$ span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

### SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3 percent to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.32}{P \times W(z)} \times K(z) \qquad \text{for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.32}{W(z) \times 0.5} \times K(z) \qquad \text{for } P \leq 0.5$$

where $F_Q^M(z)$ is measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, $F_Q$ limit is the $F_Q$ limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.14.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10 percent or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 effective full power days, whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[ \frac{F_Q^M(z)}{K(z)} \right]$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[ \frac{F_Q^M(z)}{K(z)} \right] \quad \text{is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left( \underset{\text{over } z}{\text{maximum}} \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.32}{P} \times K(z)} \right] \right) -1 \right\} \times 100 \qquad \text{for } P \geq 0.5$$

$$\left\{ \left( \underset{\text{over } z}{\text{maximum}} \left[ \frac{F_Q^M(z) \times W(z)}{\frac{2.32}{0.5} \times K(z)} \right] \right) -1 \right\} \times 100 \qquad \text{for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Place the core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied. Power level may then be increased provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limit, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

SURVEILLANCE REQUIREMENTS (Continued)

g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 15 percent inclusive.

2. Upper core region 85 to 100 percent inclusive.

4.2.2.3 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3 percent to account for manufacturing tolerances or further increased by 5 percent to account for measurement uncertainty.

Figure 3.2-2

K(Z) AS A FUNCTION OF CORE HEIGHT

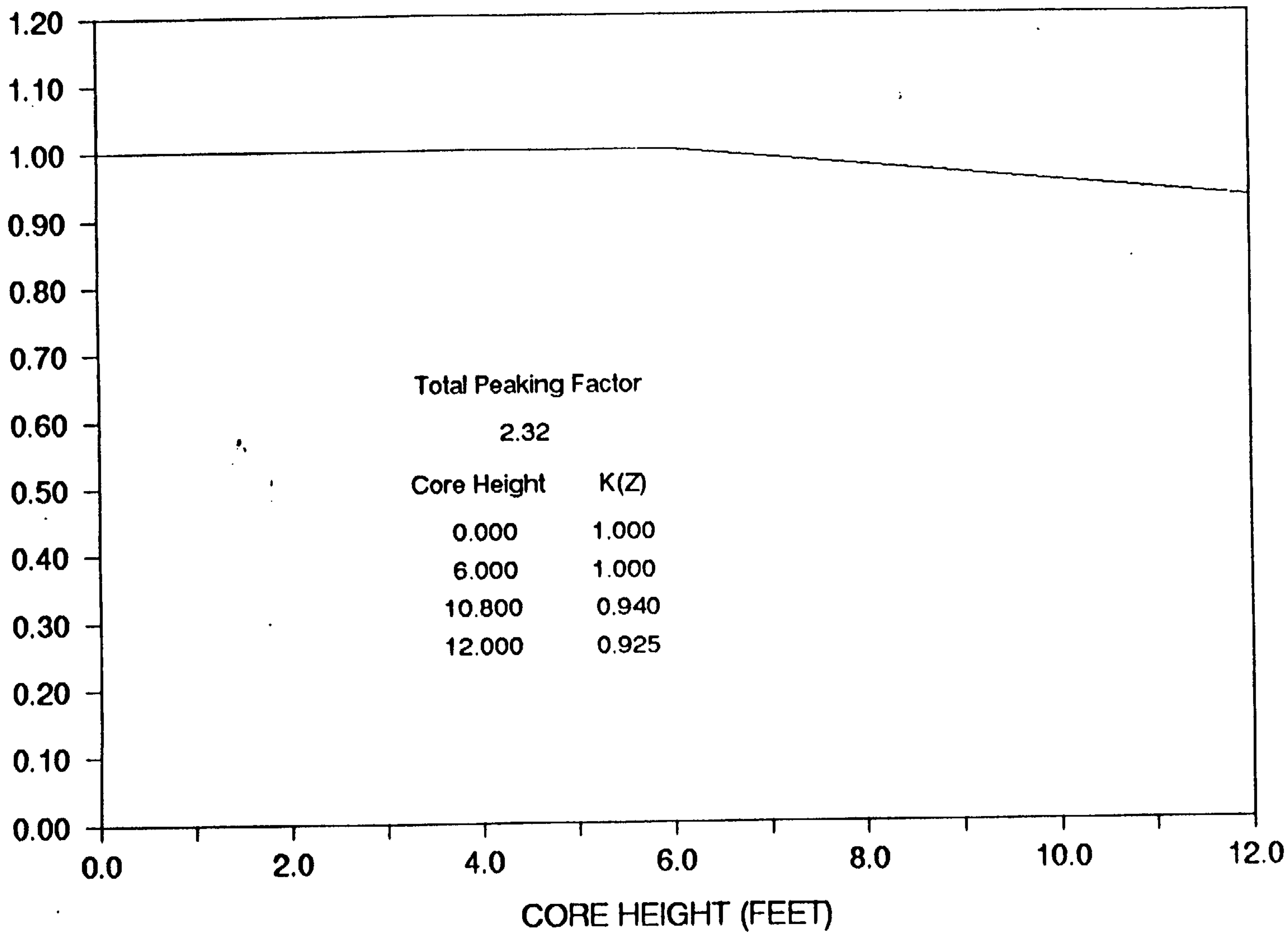


Figure 3.2-2
K(Z) - Normalized Fq(Z) as a Function of Core Height

## TABLE 3.4-1

## REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

| VALVE NUMBER | FUNCTION |
|---|---|
| 63-560 | Accumulator Discharge |
| 63-561 | Accumulator Discharge |
| 63-562 | Accumulator Discharge |
| 63-563 | Accumulator Discharge |
| 63-622 | Accumulator Discharge |
| 63-623 | Accumulator Discharge |
| 63-624 | Accumulator Discharge |
| 63-625 | Accumulator Discharge |
| 63-551 | Safety Injection (Cold Leg) |
| 63-553 | Safety Injection (Cold Leg) |
| 63-557 | Safety Injection (Cold Leg) |
| 63-555 | Safety Injection (Cold Leg) |
| 63-632 | Residual Heat Removal (Cold Leg) |
| 63-633 | Residual Heat Removal (Cold Leg) |
| 63-634 | Residual Heat Removal (Cold Leg) |
| 63-635 | Residual Heat Removal (Cold Leg) |
| 63-641 | Residual Heat Removal/Safety Injection (Hot Leg) |
| 63-644 | Residual Heat Removal/Safety Injection (Hot Leg) |
| 63-558 | Safety Injection (Hot Leg) |
| 63-559 | Safety Injection (Hot Leg) |
| 63-543 | Safety Injection (Hot Leg) |
| 63-545 | Safety Injection (Hot Leg) |
| 63-547 | Safety Injection (Hot Leg) |
| 63-549 | Safety Injection (Hot Leg) |
| 63-640 | Residual Heat Removal (Hot Leg) |
| 63-643 | Residual Heat Removal (Hot Leg) |
| FCV-74-1* | Residual Heat Removal |
| FCV-74-2* | Residual Heat Removal |

*These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS

### 3/4.5.1 ACCUMULATORS

### COLD LEG INJECTION ACCUMULATORS

### LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7615 and 8094 gallons of borated water,

c. Between 2400 and 2700 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 683 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

c.# With one pressure or water level channel inoperable per accumulator, return the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d.# With more than one channel (pressure or water level) inoperable per accumulator, immediately declare the affected accumulator(s) inoperable.

*Pressurizer pressure above 1000 psig.

#Actions c and d are in effect until the restart of Unit 2 from the Unit 2 Cycle 4 refueling outage.

## DELETED

### LIMITING CONDITION FOR OPERATION

This Specification is deleted.

## 3/4.5.2 ECCS SUBSYSTEMS - $T_{avg}$ Greater Than or Equal to 350°F

### LIMITING CONDITION FOR OPERATION

*3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE residual heat removal heat exchanger,

d. One OPERABLE residual heat removal pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

### SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

*NOTE: With one centrifugal charging pump inoperable, the emergency core cooling system (ECCS) may remain operable for an additional 36 hours beyond that identified in Action statement (a). This temporary change expires at 0848 on July 13, 1984.

SURVEILLANCE REQUIREMENTS (Continued)

| | Valve Number | Valve Function | Valve Position |
|---|---|---|---|
| a. | FCV-63-1 | RHR Suction from RWST | open |
| b. | FCV-63-22 | SIS Discharge to Common Piping | open |

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. DELETED.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal and automatic switchover to containment sump test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

    a) Centrifugal charging pump

    b) Safety injection pump

    c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

| | | |
|---|---|---|
| 1. | Centrifugal charging pump | Greater than or equal to 2400 psig |
| 2. | Safety Injection pump | Greater than or equal to 1407 psig |
| 3. | Residual heat removal pump | Greater than or equal to 165 psig |

g. By verifying the correct position of each mechanical stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

| Charging Pump Injection Throttle Valves | Safety Injection Cold Leg Throttle Valves | Safety Injection Hot Leg Throttle Valves |
|---|---|---|
| Valve Number | Valve Number | Valve Number |
| 1. 63 - 582 | 1. 63 - 550 | 1. 63-542 |
| 2. 63 - 583 | 2. 63 - 552 | 2. 63-544 |
| 3. 63 - 584 | 3. 63 - 554 | 3. 63-546 |
| 4. 63 - 585 | 4. 63 - 556 | 4. 63-548 |

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

1. For safety injection pump lines with a single pump running:

   a. The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 443 gpm, and

   b. The total pump flow rate is less than or equal to 675 gpm.

2. For centrifugal charging pump lines with a single pump running:

   a. The sum of the injection line flow rates, excluding the highest flow rate is greater than or equal to 309 gpm, and

   b. The total pump flow rate is less than or equal to 555 gpm.

3. For all four cold leg injection lines with a single RHR pump running a flow rate greater than or equal to 3931 gpm.

## 3/4.5.3 ECCS SUBSYSTEMS - $T_{avg}$ LESS THAN 350°F

### LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System $T_{avg}$ less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a REPORTABLE EVENT shall be prepared and submitted to the Commission pursuant to Specification 6.6.1. This report shall include a description of the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

## 3/4.5.4 DELETED

### LIMITING CONDITION FOR OPERATION

This Specification is deleted.

## 3/4.5.5 REFUELING WATER STORAGE TANK

### LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between 2500 and 2700 ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

TABLE 3.6-1 (Continued)
BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING
SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

| PENETRATION | DESCRIPTION | RELEASE LOCATION |
|---|---|---|
| X-92A,B | Hydrogen Analyzer | Auxiliary Area |
| X-93 | Accumulator Sample | Auxiliary Area |
| X-94A,B,C | Radiation Sample | Auxiliary Area |
| X-95A,B,C | Radiation Sample | Auxiliary Area |
| X-96C | Hot Leg Sample | Auxiliary Area |
| X-98 | ILRT | Auxiliary Area |
| X-99 | Hydrogen Analyzer | Auxiliary Area |
| X-100 | Hydrogen Analyzer | Auxiliary Area |
| X-101 | Postaccident Sampling, Containment | Auxiliary Area |
| X-103 | Postaccident Sampling, Liquid Discharge to Containment | Auxiliary Area |
| X-106 | Postaccident Sampling, Air Discharge to Containment | Auxiliary Area |
| X-108 | Maintenance Penetration | Auxiliary Area |
| X-109 | Maintenance Penetration | Auxiliary Area |
| X-114 | Ice Condenser | Auxiliary Area |
| X-115 | Ice Condenser | Auxiliary Area |
| X-116A | Postaccident Sampling, Containment Air Sample | Auxiliary Area |

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

| VALVE NUMBER | FUNCTION | MAXIMUM ISOLATION TIME (Seconds) |
|---|---|---|
| A. PHASE "A" ISOLATION (Cont.) | | |
| 61. FCV-77-19 | RCDT and PRT to V H | 10* |
| 62. FCV-77-20 | $N_2$ to RCDT | 10* |
| 63. FCV-77-127 | Floor Sump Pump Disch | 10* |
| 64. FCV-77-128 | Floor Sump Pump Disch | 10* |
| 65. FCV-81-12 | Primary Water Makeup | 10* |
| B. PHASE "B" ISOLATION | | |
| 1. FCV-32-81 | Control Air Supply | 10 |
| 2. FCV-32-103 | Control Air Supply | 10 |
| 3. FCV-32-111 | Control Air Supply | 10 |
| 4. FCV-67-83 | ERCW - LWR Cmpt Clrs | 60* |
| 5. FCV-67-87 | ERCW - LWR Cmpt Clrs | 60* |
| 6. FCV-67-88 | ERCW - LWR Cmpt Clrs | 60* |
| 7. FCV-67-89** | ERCW - LWR Cmpt Clrs | 70* |
| 8. FCV-67-90** | ERCW - LWR Cmpt Clrs | 70* |
| 9. FCV-67-91 | ERCW - LWR Cmpt Clrs | 60* |
| 10. FCV-67-95 | ERCW - LWR Cmpt Clrs | 60* |
| 11. FCV-67-96 | ERCW - LWR Cmpt Clrs | 60* |
| 12. FCV-67-99 | ERCW - LWR Cmpt Clrs | 60* |
| 13. FCV-67-103 | ERCW - LWR Cmpt Clrs | 60* |
| 14. FCV-67-104 | ERCW - LWR Cmpt Clrs | 60* |
| 15. FCV-67-105** | ERCW - LWR Cmpt Clrs | 70* |
| 16. FCV-67-106** | ERCW - LWR Cmpt Clrs | 70* |
| 17. FCV-67-107 | ERCW - LWR Cmpt Clrs | 60* |
| 18. FCV-67-111 | ERCW - LWR Cmpt Clrs | 60* |
| 19. FCV-67-112 | ERCW - LWR Cmpt Clrs | 60* |
| 20. FCV-67-130 | ERCW - Up Cmpt Clrs | 60* |
| 21. FCV-67-131 | ERCW - Up Cmpt Clrs | 60* |
| 22. FCV-67-133 | ERCW - Up Cmpt Clrs | 60* |

BORATION SYSTEMS (Continued)

provide a SHUTDOWN MARGIN from expected operating conditions of 1.6% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6042 gallons of 20,000 ppm borated water from the boric acid storage tanks or 82,082 gallons of 2500 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% delta k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks or 9,690 gallons of 2500 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the calculated DNBR in the core at or above design during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

### 3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The compuer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed $\Delta$I-Power operating space and the THERMAL POWER is greater than 50 percent of RATED THERMAL POWER.

### 3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS

The limits on heat flux hot channel factor and nuclear enthalpy hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

### 3/4.5.1 ACCUMULATORS

The OPERABILITY of each cold leg injection accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core in the event the RCS pressure falls below the pressure of the accumulators. For the cold leg injection accumulators this condition occurs in the event of a large or small rupture.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The limits in the specification for accumulator volume and nitrogen cover pressure are analysis limits and do not include instrument uncertainty. The cover pressure limits were determined by Westinghouse to be 615 psia and 697.5 psia. Since the instrument read-outs in the control room are in psig, the TS values have been converted to psig and rounded to the nearest whole numbers. The actual nitrogen cover pressure safety limits in SQN's design documents are 600.3 psig and 682.8 psig. The minimum boron concentration ensures that the reactor core will remain subcritical during the accumulator injection period of a small break LOCA.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

### 3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

## 3/4.5.4 BORON INJECTION SYSTEM

This Specification was deleted.

## 3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses. Additionally, the OPERABILITY of the RWST, as part of the ECCS, ensures that sufficent negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

ENCLOSURE

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-79

## TENNESSEE VALLEY AUTHORITY

## SEQUOYAH NUCLEAR PLANT, UNIT 2

## DOCKET NO. 50-328

## 1.0 INTRODUCTION

By the two applications dated January 12, 1990, the Tennessee Valley Authority (TVA) proposed to modify the Sequoyah Nuclear Plant, Unit 2, Technical Specifications (TSs) to remove the upper head injection system (UHIS) and to deactivate the boron injection tank (BIT). The modifications were done at Unit 1 in the Unit 1 Cycle 4 refueling outage. The modifications to Unit 2 are being done in the current Unit 2 Cycle 4 refueling outage. This evaluation will address only the changes to the Unit 2 TSs. The changes to the Unit 1 TSs were issued in the staff's letter dated May 11, 1990.

For the UHIS removal, TVA proposed the following changes to the TSs: TS 3/4.5.1.2 on the UHIS would be deleted; Tables 3.4-1, 3.6-1, and 3.6-2 for the reactor coolant system (RCS) pressure isolation valves, penetrations, and containment isolation valves would be revised; Limiting Condition for Operation (LCO) 3.2.2 and Surveillance Requirement (SR) 4.2.2.2 would be revised in terms of the peaking factor limit and, as a result of the peaking factor limit revision, Figure 3.2-2 would be revised; LCO 3.5.1.1 for the cold leg injection accumulators would reflect new values for the volume of water and nitrogen cover pressure; and SR 4.5.2.h would be revised for new minimum flow rate values for emergency core cooling systems pumps. This is TVA TS Change Request 89-25.

For the BIT deactivation, TVA proposed the following changes to the TSs: the refueling water storage tank boron concentration would be changed in LCOs 3.1.2.5 and 3.5.5, the volume of the boric acid storage system and the boron concentration of the refueling water storage tank would be changed in LCO 3.1.2.6, the reference to boron injection throttle valves will be changed to charging pump injection throttle valves in SR 4.5.2.g, TSs 3/4.5.4.1 and 3/4.5.4.2 for the boron injection system would be deleted, and the boron concentration for the cold leg injection accumulators would be changed in LCO 3.5.1.1. This is TVA TS Change Request 89-26.

There are also proposed changes to the bases of the affected TSs listed above and to the index of the TSs.

## 2.0 EVALUATION

The evaluations of the proposed changes to the TSs will be given below in Sections 2.1 and 2.2, for the removal of the UHIS and the deactivation of the BIT, respectively.

### 2.1 UHIS Removal

The UHIS has been the subject of regulatory concerns at Sequoyah (SQN), including an integrated design inspection and two licensee event reports. The system was designed to provide additional core cooling during reactor blowdown following a large break loss-of-coolant accident (LOCA). It was also modeled into the secondary side depressurization transients. Experience has demonstrated that the UHIS adds to the complexity of plant operation, requires additional maintenance and generally reduces plant availability. Because of this, plants such as McGuire and Catawba, and now Sequoyah, have proposed removal of the UHIS

By letter dated November 3, 1988, TVA committed to remove the UHIS before restart from the Unit 1 and Unit 2 Cycle 4 refueling outages. In a follow-up letter dated January 12, 1990, TVA proposed amendments to the TSs which would reflect the removal of the UHIS.

To support the request for UHIS removal, TVA has reanalyzed the following postulated events without credit for core cooling from the UHIS: (1) large and small break LOCA, (2) transients for a steamline break, and (3) the largest single failed-open steam generator relief, safety, or dump valve. In performing these analyses, TVA has considered the effects of the following modifications which were implemented during the Cycle 4 refueling outages for Units 1 and 2:

- Implementation of the Eagle 21 digital protection system
- Deactivation of the boron injection tank (see Section 2.2 below)
- Implementation of the Vantage 5 hybrid fuel
- Use of a new steamline break protection
- Elimination of a reactor trip on steam flow/feed flow mismatch

The staff evaluation of the analyses which support the proposed facility modification for the UHIS removal and associated TS changes is described in the following sections.

#### 2.1.1 Large Break LOCA Evaluation

TVA provided the results of a large break LOCA analysis supporting the request for removal of the UHIS. In the licensee's submittal only the double end cold leg guillotine (DECLG) breaks were analyzed since they were identified previously as limiting cases that result in the highest peak cladding temperature (PCT). The DECLG break analysis was performed with a total peaking factor of 2.32, 102% of the core power of 3411 Mwt, and an assumed loss of offsite power at the beginning of the accident. The effect of varying break discharge coefficients on the peak cladding temperature was studied.

The results of the study showed that the DECLG break with a discharge coefficient of 0.6 is the worst large break LOCA case resulting in a PCT of 2001.2°F which is below the acceptance criterion of 2200°F. The analysis was performed using Westinghouse Emergency Core Cooling System (ECCS) evaluation models (Ref. 1).

In their review of the analysis, the staff found that approved analytical methods and computer codes were used to perform calculations, and that the results showed that the PCT, clad oxidation and hydrogen generation are within the acceptance criteria imposed in 10 CFR 50.46 for LOCA analysis.

TVA provided a revised table of mass and energy rates used for the containment backpressure calculation as a function of time during blowdown in the large break LOCA. Removal of the UHIS was included in the containment/LOCA analysis that was submitted by TVA in its letter dated January 12, 1990 for its TS Change Request 90-05 to reduce the frequency of weighing ice in the containment ice condenser. The peak containment pressure is 10.9 psi following the large break LOCA. This peak pressure is below the design value of 12 psi and the staff accepted the containment analysis in its letter dated March 2, 1990 approving Amendments 131 and 118 for Units 1 and 2, respectively.

2.1.2 Small Break LOCA Evaluation

The small break LOCA analysis was performed with the approved computer codes, i.e., (1) the NOTRUMP (Refs. 2 and 3) code for the calculation of the transient depressurization of the reactor coolant system, core power, water-steam mixture height and steam flow past the uncovered portion of the core and (2) the LOCTA-IV (Ref. 4) code for the PCT analysis. The analysis was done assuming 102% of the core power of 3411 Mwt and a total peaking factor of 2.7. The total peaking factor of 2.7 is conservative in comparison to the proposed TS value of 2.32. Various break sizes were assumed and the results showed that the worst break size is a 3-inch diameter break. This break size results in the highest peak cladding temperature of 2105.°F which is below the acceptance criterion of 2200°F. The staff concludes that the small break LOCA analysis is acceptable since the approved method was used to show the analytical results to be within the acceptance criteria in 10 CFR 50.46.

2.1.3 Transient Evaluation

TVA used the approved LOFTRAN code (Ref. 5) to reanalyze two plant transients which were (1) a steamline break and (2) the largest single failed-open steam generator relief, safety, or dump valve. The THINC code (Refs. 6 and 7) was used to determine if departure from nucleate boiling (DNB) occurred in the core for the steamline break. For the failed open valve transient, the results of the LOFTRAN analysis were evaluated to determine if DNB occurred. The results confirmed that no DNB occurred for either the steamline break or the failed open valve and thus assured no fuel damage resulting from the transients. The THINC code has been used in prior Final Safety Analysis and is, therefore, acceptable.

The staff concludes that the licensee's transient analysis is adequate and acceptable since an approved method was used and TVA demonstrated that specified acceptable fuel design limits would not be exceeded.

2.1.4 Technical Specification Changes

The following is a list of the proposed changes to the TSs associated with the removal of the UHIS in the application dated January 12, 1990.

a. TS 3/4.2.2, Heat Flux Hot Channel Factor-$F_Q(Z)$

The proposed change would revise the total peaking factor from 2.15 to 2.32 and also replace Figure 3.2-2, the K(z), i.e., the normalized $F_Q(Z)$ curve.

The staff finds this acceptable since the ECCS analysis was performed using approved methods and gave acceptable results for the higher total peaking factor.

b. TS 3/4.4.6.2, Table 3.4-1, Reactor Coolant System Pressure Isolation Valves

UHIS valves identified as 87-558, 87-559, 87-560, 87-561, 87-562, 87-563, FCV-87-7, and FCV-87-8 were deleted from the table.

Because the removal of the UHIS results in the deletion of the UHIS reactor coolant system pressure isolation valves, this change is acceptable.

c. TS 3.5.1.1, Cold Leg Injection Accumulators

The proposed change revises the operable range of water volume between 7615 and 8094 gallons and increases the operable range of nitrogen cover-pressure between 600 and 683 psig.

The changes are consistent with the assumptions of the LOCA analysis supporting the request for removal of the UHIS. The changes are acceptable.

d. TS 3/4.5.1.2, ECCS, Upper Head Injection

The specifications associated with the operability and maintenance of the UHIS are being deleted because this system is to be removed before the restart from the current Cycle 4 refueling outage. This is acceptable.

e. TS 3/4.6.1, Containment Penetration Valves, Table 3.6-1

The proposed changes reflect the sealing of one UHIS penetration and the reclassification of the remaining UHIS bypass leakage paths to maintenance penetrations. Making these changes does not affect the requirements on containment integrity and containment leakage in TS 3/4.6.1.1 and TS 3/4.6.1.2. These changes reflect the removal of the UHIS system and are, therefore, acceptable.

f. TS 3/4.6.3, Containment Isolation Valves, Table 3.6-2

The proposed change reflects the removal of containment isolation valves associated with the UHIS containment penetration. These changes are acceptable since the UHIS system is to be removed.

g. TS 3/4.5.2, ECCS Subsystems, Tavg Greater Than or Equal to 350°F

The minimum value for the sum of the centrifugal charging pump line flow rates decreases from 316 to 309 gallons per minute.

The minimum flow rate for all four cold leg injection lines decreases from 3976 to 3931 gallons per minute.

The changes are consistent with the results of the LOCA analysis supporting the request for removal of the UHIS and are, therefore, acceptable.

h. TS Bases

TVA also proposed a change to the TS bases. The staff finds that the change to the bases of the TSs are consistent with the proposed changes to the TSs and is, therefore, acceptable.

2.1.5 UHIS Removal Conclusions

The staff has evaluated TVA's request to remove the UHIS and change the associated TSs. Based on its review of the results of the LOCA and transient analyses provided by TVA, the staff has concluded that there is reasonable assurance that the ECCS without the UHIS satisfies the performance requirements of 10 CFR 50.46 for Sequoyah. The staff, therefore, concludes that operation without the UHIS poses no undue risk to the public health and safety and is acceptable. The TS changes relating to the UHIS removal are consistent with the analytical results and the removal of the UHIS and thus are acceptable.

2.1.6 References

1. "The 1981 Version of the Westinghouse ECCS Evaluation Model Using BASH", WCAP-11524-A, Revision 2 (Non-proprietary), March 1987.

2. "NOTRUMP, A Nodal Transient Small Break and General Network Code", WCAP-10080-A, August 1985.

3. "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Codes," WCAP-10081-A, August 1985.

4. "LOCTA-IV Program: Loss-of Coolant Transient Analysis," WCAP-8305, (Non-Proprietary), WCAP-8301 (Proprietary) June, 1974.

5. "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

6. "Application of the THINC Program to PWR Design," WCAP-7359-L, August, 1969, (Proprietary), WCAP-7838, January, 1972.

7. "Application of the THINC IV Program to PWR Design," WCAP-8054, October, 1973, (Proprietary), WCAP-8195, October, 1973.


## 2.2 BIT Deactivation

In a letter dated January 12, 1990, TVA proposed changes to the TSs for the deactivation of the BIT and the deactivation or removal of its heat tracing. The BIT and piping from the charging pump to the reactor coolant is not being removed but the tank will not contain a high concentration of boron as now required by TS 3/4.5.4.1. The heat tracing as required by TS 3/4.5.4.2 will be removed or deactivated. The tank and piping will continue to serve as part of the high head/low flow emergency core cooling injection path to the reactor coolant system using the charging pumps but the boron concentration will be that of the charging flow and no greater than the concentration of boron in the borated water in the refueling water storage tank. The BIT bypass line is also being removed in response to the concerns addressed in NRC Bulletin 88-08 on thermal stresses in piping connected to the reactor coolant system. The tank inlet and discharge double isolation valves remain. These valves will continue to be normally closed and to be opened on a safety injection signal. These valves will ensure no cold water slug from the normal charging system will thermally stress any piping connected to the reactor coolant system. The tank and its associated piping will be filled and vented prior to being declared operable for power operation and are isolated from sources of noncondensables.

In performing the evaluation for removing the BIT function, TVA also considered the effects of the following planned modifications during the Cycle 4 refueling outages for Units 1 and 2:

- ° Elimination of the resistance temperature detector bypass
- ° Implementation of the Eagle 21 digital protection system
- ° Removal of the upper head injection (see Section 2.1 above)
- ° Implementation of the Vantage 5 hybrid fuel
- ° Use of a new steamline break protection
- ° Elimination of the reactor trip on steam flow/feed flow mismatch

The staff's evaluation of these proposed TS changes for deactivating the BIT is described in the following section.

## 2.2.1 Evaluation

The BITs were originally incorporated into Westinghouse-designed plants as a means of mitigating the consequences of accidental depressurization of the main steam system. The sole purpose of the BIT, as a component of the safety injection system, is to insert concentrated boric acid (i.e., 20,000 ppm) into the reactor vessel and thus create a negative reactivity during accidents. Problems and safety concerns associated with the BIT were identified in NRC Generic Letter 85-16. The high concentration of boric acid imposes operational and maintenance problems that adversely affect plant availability such as (1) minimum volumes and concentrations in boric acid system tanks, (2) heat tracing malfunctions, (3) BIT valve testing, and (4) recovery from an inadvertent safety injection. The high boric acid concentrations also cause a safety concern involving boric acid solidification that renders emergency core cooling inoperable. Therefore, many plants such as Beaver Valley, Byron/Braidwood, Turkey Point, McGuire, and Catawba have removed the BIT. TVA has decided to deactivate the BIT and the associated heat-tracing systems from SQN Units 1 and 2 during the Cycle 4 refueling outages for each unit.

TVA performed safety analyses for (1) a steamline break, with or without offsite power available, for the largest double-ended rupture of a steam pipe upstream and downstream of a flow restrictor, and (2) the largest single failed-open steam generator relief, safety, or dump valve with or without offsite power available. The staff's acceptance criterion for a main steamline break is that the radiological release should not exceed the limits set forth in 10 CFR Part 100.

The stuck open relief valve analysis is an event in which the plant may return to criticality with the acceptance criterion being that the specified acceptable fuel design limits should not be violated.

The analyses were performed by TVA using the NRC-approved method and the computer code LOFTRAN. To minimize future TS changes, TVA selected the highest possible boron concentration that would (1) accommodate the removal of the BIT, (2) accommodate removal of the upper head injection, (3) meet the requirements for the post-loss of coolant accident sump potential hydrogen-ion activity, and (4) provide the maximum available margin for future reloads. The BIT was assumed in the analysis to be at a zero ppm concentration without heat tracing. As boron was injected from the refueling water storage tank, the BIT acted in the analysis as a dilution volume, reducing the effectiveness of the boron in the refueling water storage tank (RWST).

The heat tracing for the BIT was used only to keep the temperature of the water in the BIT high enough to keep the high concentration of boron in solution. With the BIT boron concentration being reduced to at or below the proposed concentration in the refueling water storage tank, no heat tracing is needed for the BIT.

TVA stated that the design basis for the departure from the nucleate boiling ratio would be met for all cases analyzed. No fuel failures were predicted. Thus, the releases resulting from the stuck open relief valve analysis would comply with the 10 CFR Part 20 criteria. Even for the ANS Condition IV main steamline break event, using the final safety analysis report criteria of a conservative fuel failure level of 1 percent, the radiological consequence complies with 10 CFR Part 100 criteria.

TVA stated that no valves in the safety injection line through the BIT are being replaced. There are also no changes needed to be made to the engineered safety feature response times specified in the TSs due to the deactivation of the BIT.

### 2.2.2 Technical Specification Changes

The following are the proposed changes to the TSs associated with the deactivation of the BIT and the removal of its associated heat tracing.

a. Index, TS 4.5.2.g.2 and TS 3/4.5.4

The proposed changes are editorial and reflect the deactivation of the BIT and the deactivation or removal of its heat tracing.

b. TS 3.1.2.5.b.2 - RWST, Modes 5 and 6

The minimum boron concentration is increased from 2000 to 2500 ppm.

c. TS 3.1.2.6.b.2 and TS 3.5.5.b - RWST, Modes 1,2,3, and 4

The boron concentration range of 2000 to 2100 ppm is increased to a range of 2500 to 2700 ppm.

d. TS 3.1.2.6.a.1 - Boric Acid Storage System

The minimum volume of the borated water is increased from 6542 to 7176 gallons.

e. TS 3.5.1.1.c - Cold-Leg Injection Accumulators

The boron concentration is increased from a range of 1900 to 2100 ppm to a range of 2400 to 2700 ppm.

f. TS Bases

TVA also proposed a change to the TS bases. The change is consistent with the other proposed changes to the TSs and the deactivation of the BIT.

### 2.2.3 BIT Deactivation Conclusions

The staff has reviewed TVA's justification for deactivation of the BIT, deactivation or removal of its associated heat tracing, and the proposed TS changes. As NRC-approved methods were used for the analysis of the BIT deactivation and the results conform to the acceptance criteria, the proposed TS changes are acceptable.

## 2.3 Conclusions

Based on the staff's review of the two applications dated January 12, 1990 for changes to the Unit 2 TSs to reflect (1) the removal of the UHIS and (2) deactivation of the BIT at Unit 2 in the current Unit 2 Cycle 4 refueling outage, the staff concludes that these two actions and the proposed TS changes associated with these actions are acceptable. The proposed changes for the Unit 1 TSs were issued in the staff's letter dated May 11, 1990 during the Unit 1 Cycle 4 refueling outage.

The other modifications being proposed for the Unit 2 Cycle 4 refueling outage are being reviewed separately by the staff; the staff approval of the UHIS removal and the BIT deactivation does not in itself constitute acceptance of these other modifications.

## 3.0 ENVIRONMENTAL CONSIDERATION

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

## 4.0 CONCLUSION

The Commission made proposed determinations that the amendment based on TVA's applications TS 89-25 and TS 89-26 involves no significant hazards consideration. These determinations for TS 89-25 and 89-26 were published in the Federal Register (55 FR 4279 and 55 FR 4280, respectively) on February 7, 1990. The Commission consulted with the State of Tennessee. No public comments were received and the State of Tennessee did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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