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

May 5, 1989

Docket Nos. 50-327/328

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: CORRECTING INCONSISTENCIES, MINOR DISCREPANCIES AND TYPOGRAPHICAL ERRORS (TAC R00187/R00188) (TS 87-17) - SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 114 to Facility Operating License No. DPR-77 and Amendment No. 104 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated June 24, 1987.

The amendments revise the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TS). The changes are throughout the TS to correct thirty inconsistencies, minor discrepancies, factual errors and typographical errors within the TS. One change removes an error from a previous TS amendment. Twelve changes correct typographical errors. Four changes correct references to figures or the figure itself. Eight changes correct inconsistencies between the Unit 1 TS and the Unit 2 TS. Five changes correct factual errors in the TS. These corrections will eliminate confusion over applicable requirements and the potential for error in reading the TS.

The proposed change to correct the alphabetical listing of the definitions in the index was approved in Amendment 71 for Unit 1 and Amendment 63 for Unit 2. These amendments were issued by letter dated May 18, 1988.

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1/1 CP

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 114 to License No. DPR-77
2. Amendment No. 104 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 24, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 114, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 114

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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
2-7	2-7
2-9	2-9
B 2-1	B 2-1
B 2-2	B 2-2*
3/4 1-13	3/4 1-13*
3/4 1-14	3/4 1-14
3/4 1-21	3/4 1-21
3/4 3-13	3/4 3-13
3/4 3-56	3/4 3-56
3/4 3-73	3/4 3-73
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2*
3/4 7-5	3/4 7-5*
3/4 7-6	3/4 7-6
3/4 7-9	3/4 7-9*
3/4 7-10	3/4 7-10
3/4 7-37	3/4 7-37
3/4 8-4	3/4 8-4
3/4 8-5	3/4 8-5
3/4 8-6	3/4 8-6*
3/4 11-12	3/4 11-12
3/4 12-1	3/4 12-1
3/4 12-2	3/4 12-2
3/4 12-10	3/4 12-10
B 3/4 6-3	B 3/4 6-3
B 3/4 6-3a	- - -
B 3/4 6-4	B 3/4 6-4*
5-1	5-1*
5-2	5-2
5-5	5-5
5-6	5-6

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	≤ 10% Turbine Impulse Pressure Equivalent	≤ 11% Turbine Impulse Pressure Equivalent
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	≤ 35% of RATED THERMAL POWER	≤ 36% of RATED THERMAL POWER
23. Power Range Neutron Flux - (P-10) - Enable Block of Source, Intermediate, and Power Range (low setpoint) Reactor Trips	≥ 10% of RATED THERMAL POWER	≥ 9% of RATED THERMAL POWER
24. Reactor Trip P-4	Not Applicable	Not Applicable
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	≤ 50% of RATED THERMAL POWER	≤ 51% of RATED THERMAL POWER

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1}{1+\tau_1 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \left(\frac{1+\tau_2 S}{1+\tau_3 S}\right)\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right] + K_3(P-P') - f_1(\Delta I)\right\}$

where: $\frac{1}{1+\tau_1 S}$ = Lag compensator on measured ΔT

τ_1 = Time constants utilized in the lag compensator for ΔT, τ_1 = 2 secs.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15

K_2 = 0.011

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -29 percent, the ΔT trip setpoint shall be automatically reduced by 1.50 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by 0.86 percent of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right) \left(\frac{1}{1 + \tau_4 S}\right) T - K_6[T\left(\frac{1}{1 + \tau_4 S}\right) - T''] - f_2(\Delta I)\}$

Where: $\frac{1}{1 + \tau_1 S}$ = as defined in Note 1

τ_1 = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_5 S}{1 + \tau_5 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (Delta I) function of the Overtemperature Delta T trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature Delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature when it is the source of borated water.

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements,

   2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limit of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.0.

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Compare incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference greater than or equal to 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS. The test shall independently verify the OPERABILITY of the undervoltage and automatic shunt trip circuits.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Logic only, each startup or when required with the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal if not performed in previous 92 days.

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the operability of the undervoltage and shunt trip circuits for the manual reactor trip function.

(10) - Local manual shunt trip prior to placing breaker in service. Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(11) - Automatic and manual undervoltage trip.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant T_{Hot} (Wide Range)	2	1
2. Reactor Coolant T_{Cold} (Wide Range)	2	1
3. Containment Pressure (Wide Range)	2	1
4. Refueling Water Storage Tank Level	2	1
5. Reactor Coolant Pressure (Wide Range)	2	1
6. Pressurizer Level (Wide Range)	2	1
7. Steam Line Pressure	2/steam line	1/steam line
8. Steam Generator Level - (Wide Range)	1/steam generator	1/steam generator
9. Steam Generator Level - (Narrow Range)	1/steam generator	1/steam generator
10. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	1	0
12. Pressurizer PORV Position Indicator*	2/valve#	1/valve
13. Pressurizer PORV Block Valve Position Indicator**	2/valve	1/valve
14. Safety Valve Position Indicator	2/valve#	1/valve
°15. Containment Water Level (Wide Range)	2	1
16. In Core Thermocouples	4/core quadrant	2/core quadrant
17. Reactor Vessel Level Instrumentation System***	2	1

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position with power to the valve operator removed.
***This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737.
#At least one channel shall be the acoustic monitors.

TABLE 4.3-8 (Continued)

TABLE NOTATION

* During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

(5) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions occur:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.

The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room annunciation occurs if the following condition occurs:

1. Downscale failure.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-4.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint trip is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	43

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps 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. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5 each auxiliary feedwater pump shall be demonstrated OPERABLE by :

a. Verifying that:

1. each motor-driven pump develops a differential pressure of greater than or equal to 1397 psid on recirculation flow.

2. the steam-turbine driven pump develops a differential pressure of greater than or equal to 1183 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

3. at least once per 31 days, each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

b. At least once per 18 months during shutdown* by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal and a low auxiliary feedwater pump suction pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

c. At least once per 7 days by verifying that each non-automatic valve in the auxiliary feedwater system flowpath is in its correct position.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven Auxiliary Feedwater Pump.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed;

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.6.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the station to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-10 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months by:

  1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

  2. Removing the hose for inspection and re-racking, and

  3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

  1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

  2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

SURVEILLANCE REQUIREMENTS (Continued)

4. Simulating a loss of offsite power by itself, and:

    a) Verifying de-energization of the shutdown boards and load shedding from the shutdown boards.

    b) Verifying the diesel starts on the auto-start signal, energizes the shutdown boards with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the shutdown boards shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on a ESF actuation test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and

    a) Verifying de-energization of the shutdown boards and load shedding from the shutdown boards.

    b) Verifying the diesel starts on the auto-start signal, energizes the shutdown boards with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 6900 ± 690 volts and 60 ± 1.2 Hz during this test.

    c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the shutdown board and/or safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4840 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4400 kw.

The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2.d.4.b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 4400 kW.

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its shutdown status.

10. Verifying that the automatic load sequence timers are OPERABLE with the setpoint for each sequence timer within ± 5 percent of its design setpoint.

11. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Engine overspeed

    b) 86 GA lockout relay

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

f. At least once per 10 years* by:

   1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypoclorite solution, and

   2. Performing a pressure test of those portions of the diesel fuel oil system design to Section III, subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

*These requirements are waived for the initial surveillance.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.3 The 125-volt D.C. distribution panel, 125-volt D.C. battery bank and associated charger for each diesel generator shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying:

   1. That the parameters in Table 4.8-1a meet the Category A limits.

   2. That the total battery terminal voltage is greater than or equal to 124-volts on float charge.

b. At least once per 92 days by:

   1. Verifying that the parameters in Table 4.8-1a meet the Category B limits,

   2. Verifying there is no visible corrosion at either terminals or connectors, or the cell to terminal connection resistance of these items is less than 150×10^{-6} ohms, and

   3. Verifying that the average electrolyte temperature of 6 connected cells is above 60 F.

c. At least once per 18 months by verifying that:

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   2. The battery to battery and terminal connections are clean, tight and coated with anti-corrosion material.

   3. The resistance of each cell to terminal connection is less than or equal to 150×10^{-6} ohms.

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2.2.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for the purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background that will be detected with 95% probability with only a 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22\mathrm{x}10^6 \cdot Y \cdot \exp(-\lambda.\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above in microcurie per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (midpoint).

It should be noted that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of a LER, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 is detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a member of the public is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a licensee event report (LER) and pursuant to Specification 6.9.1.7, identify the cause(s) of the unavailablity of samples and identify the new locations for obtaining

LIMITING CONDITION FOR OPERATION (Continued)

replacement samples in the Annual Radiological Environmental Operating Report. A revised figure(s) and table(s) for the ODCM reflecting the new location(s) shall be included in the next semiannual radioactive effluent release report pursuant to Specification 6.9.1.9.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing fresh leafy vegetables.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment 20% greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.9.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM, if samples are available. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, mail survey, telephone survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4c. shall be followed, including analysis of control samples.

3/4.6.1.8 EMERGENCY GAS TREATMENT SYSTEM (EGTS)

The OPERABILITY of the EGTS cleanup subsystem ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the accident analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

Use of the containment purge lines is restricted to only one pair (one supply line and one exhaust line) of purge system lines at a time to ensure that the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of a loss of coolant accident during purging operations. The analysis of this accident assumed purging through the largest pair of lines (a 24 inch inlet line and a 24 inch outlet line), a pre-existing iodine spike in the reactor coolant and four second valve closure times.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 CONTAINMENT COOLING FANS

The OPERABILITY of the lower containment vent coolers ensures that adequate heat removal capacity is available to provide long-term cooling following a non-LOCA event. Postaccident use of these coolers ensures containment temperatures remain within environmental qualification limits for all safety-related equipment required to remain functional.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. By letters dated March 3, 1981, and April 2, 1981, TVA will submit a report on the operating experience of the plant no later than startup after the first refueling. This information will be used to provide a basis to re-evaluate the adequacy of the purge and vent time limits.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the hydrogen mitigation system, consisting of 68 hydrogen ignitions per unit, is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are designed to mitigate the effects of an accident as described in Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", revision 2 dated November 1978.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

The operability of at least 66 of 68 ignitors in the hydrogen mitigation system will maintain an effective coverage throughout the containment. This system of ignitors will initiate combustion of any significant amount of hydrogen released after a degraded core accident. This system is to ensure burning in a controlled manner as the hydrogen is released instead of allowing it to be ignited at high concentrations by a random ignition source.

3/4.6.5 ICE CONDENSER

The requirements associated with each of the components of the ice condenser ensure that the overall system will be available to provide sufficient pressure suppression capability to limit the containment peak pressure transient to less than 12 psig during LOCA conditions.

3/4.6.5.1 ICE BED

The OPERABILITY of the ice bed ensures that the required ice inventory will 1) be distributed evenly through the containment bays, 2) contain sufficient boron to preclude dilution of the containment sump following the LOCA and 3) contain sufficient heat removal capability to condense the reactor system volume released during a LOCA. These conditions are consistent with the assumptions used in the accident analyses.

The minimum weight figure of 1200 pounds of ice per basket contains a 10% conservative allowance for ice loss through sublimation which is a factor of 10 higher than assumed for the ice condenser design. The minimum weight figure of 2,333,100 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-1.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The shield building is a reinforced concrete building of cylindrical shape, with a dome roof around a free standing steel containment and having the following design features:

a. Nominal inside diameter = 125 feet.

b. Nominal inside height = 175 feet.

c. Minimum thickness of concrete walls = 3 feet.

d. Minimum thickness of concrete roof = 2 feet.

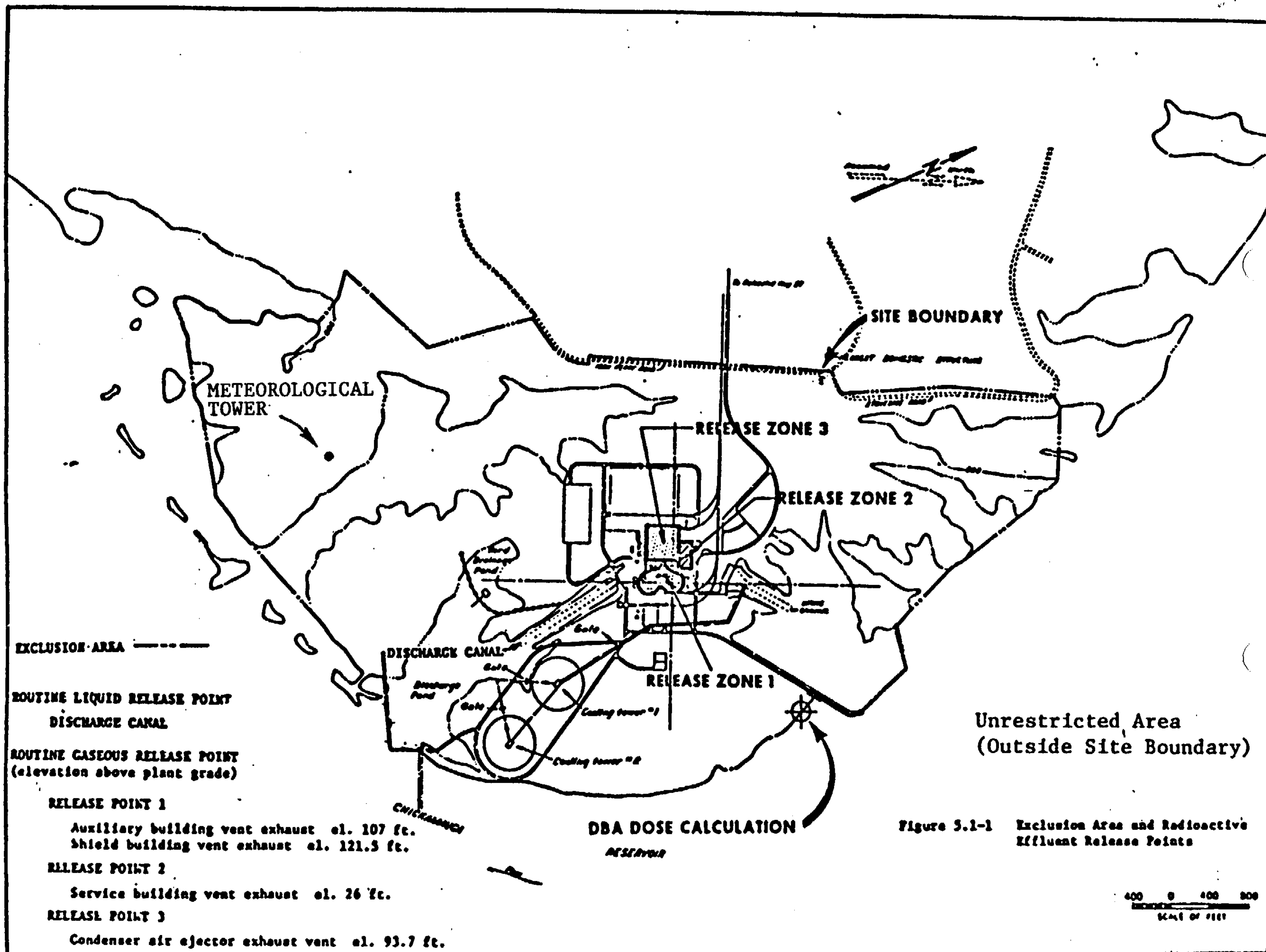
e. Minimum thickness of concrete floor pad = 9 feet.

f. Minimum thickness of steel containment = 0.5 inches at the spring line and 0.25 inches at the bottom liner plate.

g. Net free volume = 375,000 cubic feet between the steel containment and the shield building.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The steel containment is designed and shall be maintained for a maximum internal pressure of 12 psig and a temperature of 250°F.



EXCLUSION AREA

FIGURE 5.1-1

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 4.0 weight percent U-235 and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water, which includes a conservative allowance of 1.42% delta k/k for uncertainties.*

b. A nominal 10.375 inch center-to-center distance between fuel assemblies placed in the storage racks.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having an enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed. New fuel enrichment is limited to 4.0 weight percent as noted in 5.3.1 and 5.6.1.1.

DRAINAGE

5.6.2 The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

TABLE 5.7.1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/hr and 200 cooldown cycles at < 100°F/hr	Heatup cycle - T_{avg} from $\leq$ 200°F to > 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/hr	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	12 spray actuation cycles.	Spray water temperature differential > 320°F and $\leq$ 560°F.
	50 leak tests	Pressurized to 2485 psig
	5 hydrostatic pressure tests	Pressurized to 3107 psig
Secondary System	5 hydrostatic pressure tests	Pressurized to 1356 psig

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. 104
License No. DPR-79

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 24, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-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. 104, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 104

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
2-7	2-7
2-9	2-9
B 2-1	B 2-1
B 2-2	B 2-2*
3/4 1-13	3/4 1-13*
3/4 1-14	3/4 1-14
3/4 1-21	3/4 1-21
3/4 3-3	3/4 3-3*
3/4 3-4	3/4 3-4
3/4 3-7	3/4 3-7
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14*
3/4 3-57	3/4 3-57
3/4 6-4	3/4 6-4
3/4 6-18	3/4 6-18
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2*
3/4 7-5	3/4 7-5*
3/4 7-6	3/4 7-6
3/4 7-9	3/4 7-9*
3/4 7-10	3/4 7-10
3/4 7-25	3/4 7-25
3/4 7-49	3/4 7-49*
3/4 7-50	3/4 7-50
3/4 9-1	3/4 9-1
3/4 9-2	3/4 9-2*
3/4 11-9	3/4 11-9
3/4 12-2	3/4 12-2
3/4 12-9	3/4 12-9
5-1	5-1*
5-2	5-2
5-5	5-5*
5-6	5-6

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	≤ 10% Turbine Impulse Pressure Equivalent	≤ 11% Turbine Impulse Pressure Equivalent
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	≤ 35% of RATED THERMAL POWER	≤ 36% of RATED THERMAL POWER
23. Power Range Neutron Flux - (P-10) - Enable block of Source, Intermediate, and Power Range (low setpoint) reactor Trips	≥ 10% of RATED THERMAL POWER	≥ 9% of RATED THERMAL POWER
24. Reactor Trip P-4	Not Applicable	Not Applicable
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	≤ 50% of RATED THERMAL POWER	≤ 51% of RATED THERMAL POWER

NOTATION

NOTE 1: Overtemperature $\Delta T\ \left(\frac{1}{1+\tau_1 S}\right) \leq \Delta T_o\ \{K_1 - K_2\ \left(\frac{1+\tau_2 S}{1+\tau_3 S}\right)[T\left(\frac{1}{1+\tau_4 S}\right) - T'] + K_3(P-P') - f_1(\Delta I)\}$

where: $\frac{1}{1+\tau_1 S}$ = Lag compensator on measured ΔT

τ_1 = Time constants utilized in the lag compensator for ΔT, τ_1 = 2 secs.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.15

K_2 = 0.011

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

NOTE 1: (Continued)

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -29 percent, the ΔT trip setpoint shall be automatically reduced by 1.50 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by 0.86 percent of its value at RATED THERMAL POWER.

NOTE 2: Overpower $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right) \left(\frac{1}{1 + \tau_4 S}\right) T - K_6[T\left(\frac{1}{1 + \tau_4 S}\right) - T''] - f_2(\Delta I)\}$

Where:

$\frac{1}{1 + \tau_1 S}$ = as defined in Note 1

τ_1 = as defined in Note 1

ΔT_o = as defined in Note 1

K_4 $\leq$ 1.087

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

$\frac{\tau_5 S}{1 + \tau_5 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature Delta T trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid storage system solution temperature when it is the source of borated water.

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: Modes 1* and 2*.

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limit of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

*See Special Test Exceptions 3.10.2 and 3.10.3.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: Modes 1* and 2*#.

ACTION:
With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the above figure, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	$7^{\#}$
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	$7^{\#}$
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	$7^{\#}$
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	$7^{\#}$
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	$6^{\#}$
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	$6^{\#}$
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	$7^{\#}$
B. Turbine Stop Valve Closure	4	4	4	1	$13^{\#}$

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Safety Injection Input from ESF	2	1	2	1, 2	12
20. Reactor Trip Breakers					
A. Startup and Power Operation	2	1	2	1, 2	12, 15
B. Shutdown	2	1	2	3*,4* and 5*	16
21. Automatic Trip Logic					
A. Startup and Power Operation	2	1	2	1, 2	12
B. Shutdown	2	1	2	3*,4* and 5*	16
22. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	2	1	2	2, and*	8a
B. Power Range Neutron Flux, P-7	4	2	3	1	8b
C. Power Range Neutron Flux, P-8	4	2	3	1	8c
D. Power Range Neutron Flux, P-9	4	2	3	1	8e
E. Power Range Neutron Flux, P-10	4	2	3	1, 2	8d
F. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8b
G. Reactor Trip, P-4	2	1	2	1, 2, and *	14

TABLE 3.3-1 (Continued)

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, declare the interlock inoperable and verify that all affected channels of the functions listed below are OPERABLE or apply the appropriate ACTION statement(s) for those functions. Functions to be evaluated are:

a. Source Range Reactor Trip.

b. Reactor Trip

Low Reactor Coolant Loop Flow (2 loops)
Undervoltage
Underfrequency
Pressurizer Low Pressure
Pressurizer High Level

c. Reactor Trip

Low Reactor Coolant Loop Flow (1 loop)

d. Reactor Trip

Intermediate Range
Low Power Range
Source Range

e. Reactor Trip
Turbine Trip

ACTION 9 - Deleted

ACTION 10 - Deleted

ACTION 11 - Deleted

ACTION 12 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Compare incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference greater than or equal to 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS. The test shall independently verify the OPERABILITY of the undervoltage and automatic shunt trip circuits.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Logic only, each startup or when required with the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal if not performed in previous 92 days.

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the operability of the undervoltage and shunt trip circuits for the manual reactor trip function.

(10) - Local manual shunt trip prior to placing breaker in service. Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(11) - Automatic and manual undervoltage trip.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel or interlock trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel and interlock shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant T_{Hot} (Wide Range)	2	1
2. Reactor Coolant T_{Cold} (Wide Range)	2	1
3. Containment Pressure (Wide Range)	2	1
4. Refueling Water Storage Tank Level	2	1
5. Reactor Coolant Pressure (Wide Range)	2	1
6. Pressurizer Level (Wide Range)	2	1
7. Steam Line Pressure	2/steam line	1/steam line
8. Steam Generator Level - (Wide Range)	1/steam generator	1/steam generator
9. Steam Generator Level - (Narrow Range)	1/steam generator	1/steam generator
10. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	1	0
12. Pressurizer PORV Position Indicator*	2/valve#	1/valve
13. Pressurizer PORV Block Valve Position Indicator**	2/valve	1/valve
14. Safety Valve Position Indicator	2/valve#	1/valve
°15. Containment Water Level (Wide Range)	2	1
16. In Core Thermocouples	4/core quadrant	2/core quadrant
17. Reactor Vessel Level Instrumentation System***	2	1

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position with power to the valve operator removed.

***This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737.

#At least one channel shall be the acoustic monitors.

SURVEILLANCE REQUIREMENTS (Continued)

3. Valves pressurized with fluid from a seal system.

e. The combined bypass leakage rate to the auxiliary building shall be determined to be less than or equal to 0.25 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 12 psig, during each Type A test.

f. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1 3.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 13.2 psig, and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A and 49B) for at least 30 days.

h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 12 psig, at intervals no greater than once per 3 years.

i. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

j. The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve specified in Table 3.6-2 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a Containment Ventilation isolation test signal, each Containment Ventilation valve actuates to its isolation position.

d. Verifying that on a high containment pressure isolation test signal, each Containment Vacuum Relief Valve actuates to its isolation position.

e. Verifying that on a Safety Injection test signal that the Normal Charging Isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-2 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in Modes 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	65
3	43

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps 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. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5 each auxiliary feedwater pump shall be demonstrated OPERABLE by:

a. Verifying that:

   1. each motor-driven pump develops a differential pressure of greater than or equal to 1397 psid on recirculation flow.

   2. the steam-turbine driven pump develops a differential pressure of greater than or equal to 1183 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

3. at least once per 31 days, each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

b. At least once per 18 months during shutdown* by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal and a low auxiliary feedwater pump suction pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an auxiliary feedwater actuation test signal.

c. At least once per 7 days by verifying that each non-automatic valve in the auxiliary feedwater system flowpath is in its correct position.

*The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven Auxiliary Feedwater Pump.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: Modes 1, 2 and 3.

ACTION:

Modes 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

SURVEILLANCE REQUIREMENTS (Continued)

g. Functional Test Failure - Attached Component Analysis

For snubbers(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the snubber(s) were adversely affected by the inoperability of the snubbers(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

h. Functional Testing of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the funtional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not fail between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life does not expire during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance wth 6.10.2.n.

Mechanical snubber drag force increases greater than 50 percent of previously measured values shall be evaluated as an indication of impending failure of the snubber. These evaluations and any associated corrective action, shall be documented, and the documentation shall be retained in accordance with 6.10.2.n.

j. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions form the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber operability for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
a. Reactor Building - Annulus Area		
Platform	778.0	2-26-1196
Platform	778.0	2-26-1197
Platform	778.0	2-26-1198
Platform	778.0	2-26-1199
Platform	759.0	2-26-1200
Platform	759.0	2-26-1201
Platform	759.0	2-26-1202
Platform	759.0	2-26-1203
Platform	740.0	2-26-1204
Platform	740.0	2-26-1205
Platform	740.0	2-26-1206
Platform	740.0	2-26-1207
Platform	721.0	2-26-1208
Platform	721.0	2-26-1209
Platform	721.0	2-26-1210
Platform	721.0	2-26-1211
Platform	701.0	2-26-1212
Platform	701.0	2-26-1213
Platform	701.0	2-26-1214
Platform	701.0	2-26-1215
Platform	679.78	2-26-1216
Platform	679.78	2-26-1217
Platform	679.78	2-26-1218
Platform	679.78	2-26-1219
b. Reactor Building - RCP & Lower Containment Air Filters Area		
Reactor Building	679.78	2-26-1220
Reactor Building	679.78	2-26-1221
Reactor Building	679.78	2-26-1222
Reactor Building	679.78	2-26-1223
Reactor Building	679.78	2-26-1224
Reactor Building	679.78	2-26-1225
c. Control Building		
Control Building	732	0-26-1186
Control Building	732	0-26-1191
Control Building	706	0-26-1187
Control Building	706	0-26-1192

Table 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Control Building	685	0-26-1188
Control Building	685	0-26-1193
Control Building	669	0-26-1189
Control Building	669	0-26-1194
d. Diesel Generator Building		
Corridor	722	0-26-1077
Corridor	740.5	0-26-1080
Air Exhaust Rm.	740.5	0-26-1082
e. Additional Equipment Building - Unit 2		
North Wall	740.5	2-26-687
North Wall	706	2-26-686
f. Auxiliary Building		
	759	2-26-669
	749	2-26-664
	749	1-26-664
	734	2-26-670
	734	0-26-684
	734	1-26-670
	734	0-26-682
	734 Siamese Outlet	2-26-671
	734	2-26-672
	734	2-26-665
	714	0-26-660
	714	2-26-666
	714	0-26-677
	706	0-26-658
	690	0-26-690
	690	0-26-661
	690 Siamese Outlet	2-26-674
	690	2-26-675
	669	2-26-667
	669	2-26-668
	669	0-26-662
	669	0-26-680
	653	0-26-663
	653	0-26-691

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 One of the following valve combinations shall be verified closed under administrative control at least once per 72 hours:

Combination A	Combination B	Combination C	Combination D
a. 2-81-536	a. 2-81-536	a. 2-81-536	a. 2-81-536
b. 2-62-922	b. 2-62-922	b. 2-62-907	b. 2-62-907
c. 2-62-916	c. 2-62-916	c. 2-62-914	c. 2-62-914
d. 2-62-933	d. 2-62-940	d. 2-62-921	d. 2-62-921
	e. 2-62-696	e. 2-62-933	e. 2-62-940
	f. 2-62-929		f. 2-62-929
	g. 2-62-932		g. 2-62-932
	h. 2-FCV-62-128		h. 2-62-696
			i. 2-FCV-62-128

TABLE 4.11-2
RADIOACTIVE GASEOUS WASTE MONITORING SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment 1. Purge	Pi Each Purge Grab Sample	Di Each Purge	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
2. Vent	Dj Each Day Grab Sample	Dj Each Day	Principal Gamma Emitters[g] H-3	1×10^{-4} 1×10^{-6}
C. Noble Gases and Tritium 1. Condenser Vacuum Exhaust[h] 2. Auxiliary Building Exhaust[b,e] 3. Service Building Exhaust 4. Shield Building Exhaust[b,c,h]	M Grab Sample	M	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
D. Iodine and Particulates 1. Auxiliary Building Exhaust 2. Shield Building Exhaust	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}

LIMITING CONDITION FOR OPERATION (Continued)

deleted from the monitoring program. In lieu of a licensee event report (LER) and pursuant to Specification 6.9.1.7, identify the cause(s) of the unavailability of samples and identify the new locations for obtaining replacement samples in the Annual Radiological Environmental Operating Report. A revised figure(s) and table(s) for the ODCM reflecting the new location(s) shall be included in the next semiannual radioactive effluent release report pursuant to Specification 6.9.1.9.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing fresh leafy vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) which yields a calculated dose or dose commitment 20% greater than the values currently being calculated in Specification 4.11.2.3, identify the new locations(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With a Land Use Census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM, if samples are available. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, mail survey, telephone survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4c shall be followed, including analysis of control samples.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-1.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The shield building is a reinforced concrete building of cylindrical shape, with a dome roof around a free standing steel containment and having the following design features:

a. Nominal inside diameter = 125 feet.

b. Nominal inside height = 175 feet.

c. Minimum thickness of concrete walls = 3 feet.

d. Minimum thickness of concrete roof = 2 feet.

e. Minimum thickness of concrete floor pad = 9 feet.

f. Minimum thickness of steel containment liner = 0.5 inches at the spring line and 0.25 inches at the bottom liner plate.

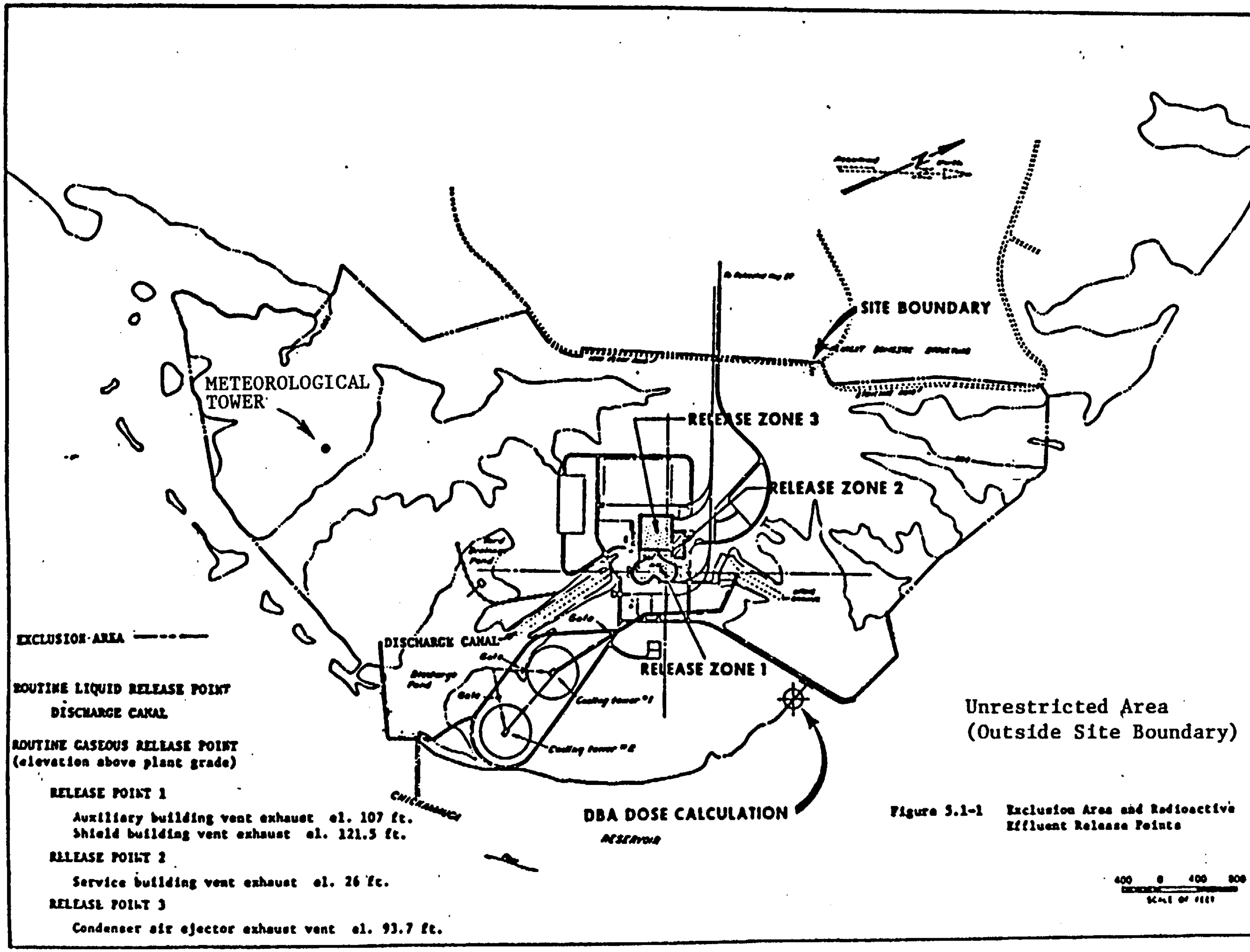
g. Net free volume = 3.75×10^5 cubic feet between the steel containment and the shield building.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The steel containment is designed and shall be maintained for a maximum internal pressure of 12 psig and a temperature of 250°F.

EXCLUSION AREA

FIGURE 5.1-1



5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed for fuel enriched to 4.0 weight percent U-235 and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water, which includes a conservative allowance of 1.42% delta k/k for uncertainties.*

b. A nominal 10.375 inch center-to-center distance between fuel assemblies placed in the storage racks.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having an enrichment of 4.5 weight percent U-235 is in place and optimum achievable moderation is assumed. New fuel enrichment is limited to 4.0 weight percent, as noted in 5.3.1 and 5.6.1.1.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

*For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident.

TABLE 5.7.1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/hr and 200 cooldown cycles at < 100°F/hr	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/hr	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate turbine or reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss of offsite A.C. electrical power.	Loss of offsite A.C. electrical power source supplying the onsite ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	12 spray actuation cycles.	Spray water temperature differential > 320°F and $\leq$ 560°F.
	50 leak tests	Pressurized to 2485 psig
	5 hydrostatic pressure tests	Pressurized to 3107 psig
Secondary System	5 hydrostatic pressure tests	Pressurized to 1356 psig

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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated June 24, 1987, the Tennessee Valley Authority (TVA) proposed 31 separate changes to the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TS). These changes are throughout the TS to correct inconsistencies, minor discrepancies, factual errors and typographical errors. There are twelve changes that would correct typographical errors, one change that would correct the alphabetical listing of the definitions, one change would correct an error from a previous amendment, four changes that would correct references to figures or the figure itself, eight changes that would correct inconsistencies between the Unit 1 and Unit 2 TS and five changes that would correct factual errors in the TS. These corrections will eliminate confusion over applicable requirements and the potential for error in reading the TS.

2.0 EVALUATION

There are 31 separate items that TVA proposes to change in the TS. There are changes that affect both Unit 1 and Unit 2, changes that affect only Unit 1 and changes that affect only Unit 2. These changes are listed in the enclosed table. The change to correct the alphabetical listing of the definitions in the index (i.e., item 1) was approved in Amendment 71 for Unit 1 and Amendment 63 for Unit 2. These amendments were issued by letter dated May 18, 1988.

The proposed changes have been reviewed by the staff. The changes are consistent with the Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 4a, and the TS, and are correct. Sequoyah is a Westinghouse pressurized water reactor. Therefore, the staff concludes that the proposed changes are acceptable.

In the Unit 1 TS, there are the following two additional inconsistency errors: (1) page B3/4 6-3a duplicates the top of page 3/4 6-4 of the Unit 1 Bases on combustible gas control and (2) the dimension "inch" was not included in the phrase "21.0 center-to-center distance between new fuel assemblies" for the new fuel pit storage racks in the Unit 1 TS Section 5.6.1.2. Deleting page B 3/4 6-3a and adding the word inch to have the expression "21.0 inch center-to-center distance" will make the Unit 1 TS consistent with the Unit 2 TS. The new fuel pit storage racks are shared by both units. This was discussed with TVA by telephone on April 18, 1989 and TVA agreed to the changes. The staff concludes that these two corrections are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve 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 amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need to be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (53 FR 13021) on April 20, 1988 and 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 nor to the health and safety of the public.

Principal Contributor: J. Donohew

Dated: May 5, 1989

TABLE

DESCRIPTION OF CHANGE TO THE TECHNICAL SPECIFICATIONS (TS)

Item	TS Page	Description of Proposed Change
Changes That Affect Unit 1 (U1) and Unit 2 (U2)		
1.	I II	Correct alphabetical listing of definition section index.
2.	2-7	Add Laplace variable to the lag compensation and correct typographical error in definition of the time constant.
3.	2-9	Correct typographical errors in definition of K_4.
4.	B2-1	Delete references to nonexistent Figure 2.1-2.(a)
5.	3/4 1-14	Delete references to nonexistent Figure 3.1-2.(a)
6.	3/4 1-21	Delete references to nonexistent Figure 3.1-2.(a)
7.	3/4 3-13	Correct typographical errors in Table 4.3-1 notation. The words being capitalized are words defined in the definition section of the TS, and, therefore, should be capitalized.
8.	3/4 3-56 (U1) 3/4 3-57 (U2)	Correctly identify the required number of channels and minimum number of channels operable for auxiliary feedwater flow in the list of accident monitoring instrumentation to be consistent with plant design.
9.	3/4 7-1	Correct an inconsistency between action b. and Specification 3.4.1.1 for operation with less than four reactor coolant pumps running.
10.	3/4 7-6	Correct an error from a previous amendment that inadvertently omitted the surveillance frequency for verification of control valve operability.(b) The correction is consistent with the Standard Technical Specifications for Westinghouse Pressurizer Water Reactors, Revision 4a.

Item	TS Page	Description of Proposed Change
Changes That Affect Unit 1 (U1) and Unit 2 (U2)		
11.	3/4 7-10	Correct an inconsistency within the action statement for Modes 1 and 3.
12.	3/4 12-1 (U1 only) 3/4 12-2	Correctly identify radiological reporting requirements of Specification 3.12.1.c.
13.	3/4 12-10 (U1) 3/4 12-9 (U2)	Correctly identify radiological reporting requirements of Specifications 3.12.2, action a, and 4.12.2.
14.	5-2	Add the location of the meteorological tower to Figure 5.1-1.
15.	5-6	Correct the hydrostatic test pressures for the reactor coolant system and secondary side in Table 5.7.1.
Changes That Affect Unit 1 Only		
16.	B2-1	Correct typographical errors in the enthalpy hot channel factor terms.
17.	3/4 3-73	Correct typographical error in page heading.
18.	3/4 7-37	Correct an incorrect reference in the limiting condition for operation.
19.	3/4 8-4	Correct an inconsistency in surveillance requirement 4.8.1.1.2.d.6.b and the corresponding requirement in the Unit 2 TS and NRC Standard Technical Specifications (STS).
20.	3/4 8-5	Correct an inconsistency in surveillance requirement 4.8.1.1.2.d.7 and the corresponding requirement in the Unit 2 TS and NRC STS.
21.	3/4 11-12	Correct an inconsistency between Table 4.11-2, note a., and the corresponding requirement in the Unit 2 TS and NRC STS.
22.	B3/4 6-3	Correct a typographical error in Item 3/4.6.1.8.

Item	TS Page	Description of Proposed Change
Changes That Affect Unit 2 Only		
23.	3/4 3-4	Correct an inconsistency between Table 3.3-1, Item 22, and the corresponding item in the Unit 1 TS and the NRC STS.
24.	3/4 3-7	Correct an inconsistency between Table 3.3-1, Action 8, and the corresponding item in the Unit 1 TS.
25.	3/4 6-4	Correct a typographical error in Item e.
26.	3/4 6-18	Correct a typographical error in Specification 4.6.3.4.
27.	3/4 7-25	Correct a typographical error in Item i.
28.	3/4 7-50	Correct a typographical error in one valve number.
29.	3/4 9-1	Correct a typographical error in the footnote.
30.	3/4 11-9	Correct inconsistencies in Table 4.11-2 and the corresponding requirement in the Unit 1 TS and NRC STS.
31.	3/4 12-2	Correct typographical error in Specification 3.12.1.c.

(a) Amendment 41 for Unit 1 and Amendment 33 for Unit 2, issued September 3, 1985, deleted Figures 2.1-2 and 3.1-2.

(b) Amendment 12 for Unit 1 issued March 25, 1982, did not include the phrase, "At least once per 31 days by verifying that," to Surveillance Requirement 4.7.1.2.a.

Item	TS Page	Description of Proposed Change
Changes That Affect Unit 2 Only		
23.	3/4 3-4	Correct an inconsistency between Table 3.3-1, Item 22, and the corresponding item in the Unit 1 TS and the NRC STS.
24.	3/4 3-7	Correct an inconsistency between Table 3.3-1, Action 8, and the corresponding item in the Unit 1 TS.
25.	3/4 6-4	Correct a typographical error in Item e.
26.	3/4 6-18	Correct a typographical error in Specification 4.6.3.4.
27.	3/4 7-25	Correct a typographical error in Item i.
28.	3/4 7-50	Correct a typographical error in one valve number.
29.	3/4 9-1	Correct a typographical error in the footnote.
30.	3/4 11-9	Correct inconsistencies in Table 4.11-2 and the corresponding requirement in the Unit 1 TS and NRC STS.
31.	3/4 12-2	Correct typographical error in Specification 3.12.1.c.

(a) Amendment 41 for Unit 1 and Amendment 33 for Unit 2, issued September 3, 1985, deleted Figures 2.1-2 and 3.1-2.

(b) Amendment 12 for Unit 1 issued March 25, 1982, did not include the phrase, "At least once per 31 days by verifying that," to Surveillance Requirement 4.7.1.2.a.