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

December 7, 1990

Docket Nos. 50-327
and 50-328

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

Dear Mr. Kingsley:

SUBJECT: POST ACCIDENT MONITORING INSTRUMENTATION (TAC NOS. 75841/75842) (TS 89-30) - SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-77 and Amendment No. 135 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. The amendments are in response to your application dated January 22, 1990.

The amendments revise Section 3/4.3.3, Monitoring Instrumentation of the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). The changes revise the requirements in TS 3/4.3.3.7, Accident Monitoring Instrumentation, by (1) deleting Table 4.3-7 and adding the requirements to Surveillance Requirement 4.3.3.7; (2) transferring the action statements for Limiting Condition for Operation 3.3.3.7 to Table 3.3-10 and revising the action statements, the total number of instruments channels available, and the minimum number of channels required; and (3) revising the instruments listed in Table 3.3-10. The position indication for the pressurizer power-operated relief and block valves, and the safety valves, and the high-range and mid-range effluent monitors for noble gases are being deleted from Table 3.3-10. The narrow-range containment pressure, the auxiliary feedwater level control valve position indication and the source-range and intermediate-range neutron flux instruments are being added to Table 3.3-10. The revisions to Table 3.3-10 also add information on the specific instrument loops associated with the individual instrument listed in the table. The instruments for Unit 1 were modified to meet the provisions of Regulatory Guide 1.97 in the Unit 1 Cycle 4 refueling outage, which was completed in May 1990. The instruments for Unit 2 were modified in the recently completed Unit 2 Cycle 4 refueling outage.

A Notice of Issuance for this amendment will be included in the Commission's biweekly Federal Register notice. A copy of the Safety Evaluation is also enclosed.

In your application, the only instrumentation that you proposed for Table 3.3-10, Accident Monitoring Instrumentation, is the Regulatory Guide (RG) 1.97 Type A, Category 1 instrumentation for Sequoyah. It is the staff's position that all RG 1.97 Category 1 instrumentation, not just the Type A, should be listed in the TSs. Therefore, we request that within 120 days of receipt of this letter you (1) withdraw your request to delete the wide-range containment pressure and

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reactor vessel level instrumentation from Table 3.3-10, (2) propose adding the following two RG 1.97 Category 1 instruments to Table 3.3-10: (a) containment isolation valve position indication and (b) the essential raw cooling water system to auxiliary feedwater pumps valve position indication, and (3) withdraw or revise your request to revise the bases of TS 3/4.3.3.7 concerning RG 1.97 Type A, Category 1 instruments. The list of RG 1.97 Category 1 instrumentation for Sequoyah is Attachment 1 to your letter dated May 7, 1990 on RG 1.97.

The amended Table 3.3-10 is in a different format than the previous Table 3.3-10. In keeping the wide-range containment pressure and reactor vessel level instrumentation systems in Table 3.3-10, as discussed in the previous paragraph, we revised the format of the data and action statements for these instruments. This was discussed and agreed upon with Sequoyah Site Licensing during my visit to the site on August 16, 1990. The format of the data and action statements were revised but the requirements on these instruments were not changed.

In your application, you also proposed action statements for inoperable subcooling margin monitors which were not consistent with the guidance in Generic Letter (GL) 83-37. These monitors were upgraded in the Cycle 4 refueling outage for each unit to (1) have two monitors for each unit, (2) meet the requirements of TMI Action Plan Item II.F.2, and (3) meet the provisions of RG 1.97. Because of these upgrades, the actions statements for these monitors should be consistent with the guidance in GL 83-37. Because the proposed action statements are not consistent with the GL and you did not provide a justification for this inconsistency, we also request that within 120 days of receipt of this letter you (1) propose action statements for inoperable subcooling margin monitor consistent with the guidance in GL 83-37 or (2) provide justification that this guidance in GL 83-37 is not applicable to Sequoyah. Because the proposed action statements do not reduce the requirements on the subcooling margin monitors they are acceptable until we resolve this issue.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance for the requests for information in this letter is not required under P.L. 17-511.

Sincerely,

JNDonohew

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

Enclosures:
1. Amendment No. 149 to License No. DPR-77
2. Amendment No. 135 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures:
See next page

Not Needed

OFC	:PDII-4/LA	:PDII-4/PM	:SRXB/BC	:SICB/BC	:OGC	:OTSB/BC:	:PDII-4/DD	:PDII-4/D
NAME	:MKrebs MK	:JDonohew	:RJones	:SNewberry	RBachmann	:JCalvo	:SBlack	:FHebdon
DATE	11/16/90	:11/8/90	:11/9/90	: 11/13/90	:11/21/90	:11/ /90	:11/14/90	:11/4/90

OFFICIAL RECORD COPY

AMENDMENT NO. 149 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-327 AND
AMENDMENT NO. 135 FOR SEQUOYAH UNIT NO. 1 - DOCKET NO. 50-328
DATED: December 7, 1990

DISTRIBUTION:

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R. Jones	8-E-23
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J. Calvo	11-F-22
ACRS (10)	
GPA/PA	2-G-5
OC/LFMB	MNBB-9112
SQN Reading File	

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Mr. Oliver D. Kingsley, Jr.

cc:
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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. 149
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 January 22, 1990, 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. 149, 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

Frederick J. Hebdon

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 7, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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
3/4 3-55	3/4 3-55
3/4 3-56	3/4 3-56
3/4 3-56a	3/4 3-56a
3/4 3-57	3/4 3-57
3/4 3-57a	3/4 3-57a
B3/4 3-3	B3/4 3-3
B3/4 3-4	B3/4 3-4
B3/4 3-5	---
B3/4 6-4	B3/4 6-4

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION: As shown in Table 3.3-10

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. Every 31 days by performance of a CHANNEL CHECK, and

b. Every 18 months by performance of a CHANNEL CALIBRATION.*

*For Containment Area Radiation Monitors, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a single calibration check of the detector below 10R/h with either an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS REQUIRED	ACTION
1. Reactor Coolant T_{Hot} (Wide Range) (Instrument Loops 68-001,-024,-043,-065)	4(1/RCS Loop)	1/RCS Loop	1
2. Reactor Coolant T_{Cold} (Wide Range) (Instrument Loops 68-018,-041,-060,-083)	4(1/RCS Loop)	1/RCS Loop	1
3. Containment Pressure (Wide Range) (Instrument Loops 30-310,-311)	2	2	1
4. Containment Pressure (Narrow Range) (Instrument Loops 30-044,-045)	2	2	1
5. Refueling Water Storage Tank Level (Instrument Loops 63-050,-051)	2	2	1
6. Reactor Coolant Pressure (Wide Range) (Instrument Loops 68-062,-066,-069)	3	3	2
7. Pressurizer Level (Wide Range) (Instrument Loops 68-320,-335,-339)	3	3	2
8. Steam Line Pressure (Instrument Loops 1-002A,-002B,-009A,-009B, -020A,-020B,-027A,-027B)	2/steam line	2/steam line	1
9. Steam Generator Level - (Wide Range) (Instrument Loops 3-043,-056,-098,-111)	1/steam generator	1/steam generator	1
10. Steam Generator Level - (Narrow Range) (Instrument Loops 3-039,-042,-052,-055, -094,-097,-107,-110)	2/steam generator	2/steam generator	1
11. Auxiliary Feedwater			
a. Flow Rate (Instrument Loops 3-163,-155,-147,-170)	1/steam generator	1/steam generator	5
b. Valve Position Indication (Instrument Loops 3-164,-164A,-172,-156, -156A,-173,-148,-148A,-174,-171,-171A,-175)	3/steam generator	3/steam generator	5

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS REQUIRED	ACTION
12. Reactor Coolant System Subcooling Margin Monitor (Instrument Loops 94-101,-102)	2	2	6
13. Containment Water Level (Wide Range) (Instrument Loops 63-178,-179)	2	2	1
14. In Core Thermocouples	65	1/core quadrant/train	3
15. Reactor Vessel Level Instrumentation System (Instrument Loops 68-367,-368, -369,-370,-371,-372)	2	2	1
16. Containment Area Radiation Monitors			
a. Upper Compartment (Instrument Loops 90-271,-272)	2	1	4
b. Lower Compartment (Instrument Loops 90-273,-274)	2	1	4
17. Neutron Flux			
a. Source Range (Instrument Loops 92-5001,-5002)	2	2#	1
b. Intermediate Range (Instrument Loops 92-5003,-5004)	2	2	1

#Source Range outputs may be disabled above the P-6 (Block of Source Range Reactor Trip) setpoint.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 1 - NOTE: Also refer to the applicable action requirements from Tables 3.3-1, 3.3-3, and 3.3-9 since they may contain more restrictive actions.

a. With the number of channels one less than the minimum channels required, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 2 - NOTE: Also refer to the applicable action requirements from Tables 3.3-1 since it may contain more restrictive actions.

a. With the number of channels one less than the minimum channels required, restore the inoperable channel to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of channels three less than the minimum channels required, restore one channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 3 - a. With the number of channels less than the minimum channels required, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS
(Continued)

ACTION 4 - a. With the number of channels less than the minimum channels required, initiate an alternate method of monitoring containment area radiation within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days, or prepare and submit a special report to the Commission pursuant to Specification 6.9.2.1 within 14 days that provides actions taken, cause of the inoperability, and plans and schedule for restoring the channels to OPERABLE status.

b. The provisions of Specification 3.0.4 are not applicable.

ACTION 5 - NOTE: Also refer to the applicable action requirements from Table 3.3-9 since it may contain more restrictive actions.

a. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 6 - a. With the number of channels less than the minimum channels required, restore the inoperable channel to OPERABLE status within 7 days or increase by one the minimum shift crew per Table 6.2-1. The additional shift crew member shall be dedicated to and capable of determining the subcooling margin during an accident using existing instrumentation.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 48 hours or increase by one the minimum shift crew per Table 6.2-1. The additional shift crew member shall be dedicated to and capable of determining the subcooling margin during an accident using existing instrumentation.

c. The provisions of Specification 3.0.4 are not applicable.

design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility and the potential capability for subsequent cold shutdown from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

This specification deleted.

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

For Sequoyah, the redundant channel capability for Auxiliary Feedwater (AFW) flow consists of a single AFW flow channel for each Steam Generator with the second channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine driven AFW flowpath) for each steam generator. Two containment hydrogen monitoring channels are designated as accident monitoring instrumentation (Type A, Category 1) in accordance with Regulatory Guide 1.97. Operability and Surveillance Requirements for the purpose of accident monitoring is governed by Specification 3.6.4.1 for containment hydrogen monitors.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9

This Specification is deleted.

3/4.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements for monitoring potentially explosive gas mixtures.

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 monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

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 1155 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,245,320 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the

UNITED STATES
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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
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 January 22, 1990, 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. 135, 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

Frederick J. Hebdon

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 7, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 135

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 and spillover pages which do not have any material being changed by this amendment are marked with an "*" and provided to maintain document completeness.

REMOVE	INSERT
3/4 3-56	3/4 3-56
3/4 3-57	3/4 3-57
3/4 3-57a	3/4 3-57a
3/4 3-58	3/4 3-58
3/4 3-58a	3/4 3-58a
B3/4 3-3	B3/4 3-3
B3/4 3-4	B3/4 3-4*
B3/4 6-3	B3/4 6-3*
B3/4 6-4	B3/4 6-4
B3/4 6-5	B3/4 6-5*
B3/4 6-6	B3/4 6-6*

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION: As shown in Table 3.3-10

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE:

a. Every 31 days by performance of a CHANNEL CHECK, and

b. Every 18 months by performance of a CHANNEL CALIBRATION.*

*For Containment Area Radiation Monitors, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a single point calibration of the detector below 10R/h with either an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS REQUIRED	ACTION
1. Reactor Coolant T_{Hot} (Wide Range) (Instrument Loops 68-001,-024,-043,-065)	4(1/RCS Loop)	1/RCS Loop	1
2. Reactor Coolant T_{Cold} (Wide Range) (Instrument Loops 68-018,-041,-060,-083)	4(1/RCS Loop)	1/RCS Loop	1
3. Containment Pressure (Wide Range) (Instrument Loops 30-310,-311)	2	2	1
4. Containment Pressure (Narrow Range) (Instrument Loops 30-044,-045)	2	2	1
5. Refueling Water Storage Tank Level (Instrument Loops 63-050,-051)	2	2	1
6. Reactor Coolant Pressure (Wide Range) (Instrument Loops 68-062,-066,-069)	3	3	2
7. Pressurizer Level (Wide Range) (Instrument Loops 68-320,-335,-339)	3	3	2
8. Steam Line Pressure (Instrument Loops 1-002A,-002B,-009A,-009B, -020A,-020B,-027A,-027B)	2/steam line	2/steam line	1
9. Steam Generator Level - (Wide Range) (Instrument Loops 3-043,-056,-098,-111)	1/steam generator	1/steam generator	1
10. Steam Generator Level - (Narrow Range) (Instrument Loops 3-039,-042,-052,-055, -094,-097,-107,-110)	2/steam generator	2/steam generator	1
11. Auxiliary Feedwater			
a. Flow Rate (Instrument Loops 3-163,-155,-147,-170)	1/steam generator	1/steam generator	5
b. Valve Position Indication (Instrument Loops 3-164,-164A,-172,-156, -156A,-173,-148,-148A,-174,-171,-171A,-175)	3/steam generator	3/steam generator	5

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS REQUIRED	ACTION
12. Reactor Coolant System Subcooling Margin Monitor (Instrument Loops 94-101,-102)	2	2	6
13. Containment Water Level (Wide Range) (Instrument Loops 63-178,-179)	2	2	1
14. In Core Thermocouples	65	1/core quadrant/train	3
15. Reactor Vessel Level Instrumentation System (Instrument Loops 68-367,-368, -369,-370,-371,-372)	2	2	1
16. Containment Area Radiation Monitors			
a. Upper Compartment (Instrument Loops 90-271,-272)	2	1	4
b. Lower Compartment (Instrument Loops 90-273,-274)	2	1	4
17. Neutron Flux			
a. Source Range (Instrument Loops 92-5001,-5002)	2	2#	1
b. Intermediate Range (Instrument Loops 92-5003,-5004)	2	2	1

#Source Range outputs may be disabled above the P-6 (Block of Source Range Reactor Trip) setpoint.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 1 - NOTE: Also refer to the applicable action requirements from Tables 3.3-1, 3.3-3, and 3.3-9 since they may contain more restrictive actions.

a. With the number of channels one less than the minimum channels required, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 2 - NOTE: Also refer to the applicable action requirements from Tables 3.3-1 since it may contain more restrictive actions.

a. With the number of channels one less than the minimum channels required, restore the inoperable channel to OPERABLE status within 31 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of channels three less than the minimum channels required, restore one channel to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 3 - a. With the number of channels less than the minimum channels required, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 -

a. With the number of channels less than the minimum channels required, initiate an alternate method of monitoring containment area radiation within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days, or prepare and submit a special report to the Commission pursuant to Specification 6.9.2.1 within 14 days that provides actions taken, cause of the inoperability, and plans and schedule for restoring the channels to OPERABLE status.

b. The provisions of Specification 3.0.4 are not applicable.

ACTION 5 - NOTE: Also refer to the applicable action requirements from Table 3.3-9 since it may contain more restrictive actions.

a. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of channels on one or more steam generators less than the minimum channels required for either flow rate or valve position, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 6 -

a. With the number of channels less than the minimum channels required, restore the inoperable channel to OPERABLE status within 7 days or increase by one the minimum shift crew per Table 6.2-1. The additional shift crew member shall be dedicated to and capable of determining the subcooling margin during an accident using existing instrumentation.

b. With the number of channels two less than the minimum channels required, restore at least one inoperable channel to OPERABLE status within 48 hours or increase by one the minimum shift crew per Table 6.2-1. The additional shift crew member shall be dedicated to and capable of determining the subcooling margin during an accident using existing instrumentation.

c. The provisions of Specification 3.0.4 are not applicable.

3/4.3.3.3 SEISMIC INSTRUMENTATION (Continued)

design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. All specified measurement ranges represent the minimum ranges of the instruments. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility and the potential capability for subsequent cold shutdown from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

This specification deleted.

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

For Sequoyah, the redundant channel capability for Auxiliary Feedwater (AFW) flow consists of a single AFW flow channel for each Steam Generator with the second channel consisting of three AFW valve position indicators (two level control valves for the motor driven AFW flowpath and one level control valve for the turbine drive AFW flowpath) for each steam generator. Two containment hydrogen monitoring channels are designated as accident monitoring instrumentation (Type A, Category 1) in accordance with Regulatory Guide 1.97. Operability and Surveillance Requirements for the purpose of accident monitoring is governed by Specification 3.6.4.1 for containment hydrogen monitors.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9

This Specification is deleted.

3/4.3.3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements for monitoring potentially explosive gas mixtures.

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.

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 igniters 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 monitors of Specification 3.6.4.1 are part of the accident monitoring instrumentation in Specification 3.3.3.7 and are designated as Type A, Category 1 in accordance with Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980.

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 igniters in the hydrogen control distributed ignition 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 1155 pounds of ice per basket contains a 15% conservative allowance for ice loss through sublimation which is a factor of 15 higher than assumed for the ice condenser design. The minimum weight figure of 2,245,320 pounds of ice also contains an additional 1% conservative allowance to account for systematic error in weighing instruments. In the event that observed sublimation rates are equal to or lower than design predictions after three years of operation, the minimum ice baskets weight may be adjusted downward. In addition, the number of ice baskets required to be weighed each 9 months may be reduced after 3 years of operation if such a reduction is supported by observed sublimation data.

3/4.6.5.2 ICE BED TEMPERATURE MONITORING SYSTEM

The OPERABILITY of the ice bed temperature monitoring system ensures that the capability is available for monitoring the ice temperature. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.3 ICE CONDENSER DOORS

The OPERABILITY of the ice condenser doors ensures that these doors will open because of the differential pressure between upper and lower containment resulting from the blowdown of reactor coolant during a LOCA and that the blowdown will be diverted through the ice condenser bays for heat removal and thus containment pressure control. The requirement that the doors be maintained closed during normal operation ensures that excessive sublimation of the ice will not occur because of warm air intrusion from the lower containment.

3/4.6.5.4 INLET DOOR POSITION MONITORING SYSTEM

The OPERABILITY of the inlet door position monitoring system ensures that the capability is available for monitoring the individual inlet door position. In the event the monitoring system is inoperable, the ACTION requirements provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

3/4.6.5.5 DIVIDER BARRIER PERSONNEL ACCESS DOORS AND EQUIPMENT HATCHES

The requirements for the divider barrier personnel access doors and equipment hatches being closed and OPERABLE ensure that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of the steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.5.6 CONTAINMENT AIR RETURN FANS

The OPERABILITY of the containment air return fans ensures that following a LOCA 1) the containment atmosphere is circulated for cooling by the spray system and 2) the accumulation of hydrogen in localized portions of the containment structure is minimized.

3/4.6.5.7 and 3/4.6.5.8 FLOOR AND REFUELING CANAL DRAINS

The OPERABILITY of the ice condenser floor and refueling canal drains ensures that following a LOCA, the water from the melted ice and containment spray system has access for drainage back to the containment lower compartment and subsequently to the sump. This condition ensures the availability of the water for long term cooling of the reactor during the post accident phase.

3/4.6.5.9 DIVIDER BARRIER SEAL

The requirement for the divider barrier seal to be OPERABLE ensures that a minimum bypass steam flow will occur from the lower to the upper containment compartments during a LOCA. This condition ensures a diversion of steam through the ice condenser bays that is consistent with the LOCA analyses.

3/4.6.6 VACUUM RELIEF VALVES

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than 0.1 psid. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of 0.5 psid.

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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-77

SUPPORTING AMENDMENT NO. 135 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 January 22, 1990, the Tennessee Valley Authority (TVA) proposed changes to Section 3/4.3.3, "Monitoring Instrumentation," of the Sequoyah Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). The proposed changes are to revise TS 3/4.3.3.7, "Accident Monitoring Instrumentation," and the associated Tables 3.3-10 and 4.3-7. These changes would revise the requirements in TS 3/4.3.3.7 in the following manner: (1) delete Table 4.3-7 and add requirements to Surveillance Requirement (SR) 4.3.3.7; (2) transfer the action statements for Limiting Condition for Operation (LCO) 3.3.3.7 to Table 3.3-10 and revise the action statements, the total number of instrument channels available, and the minimum number of channels required; (3) add information on the specific instrument loops to the instruments listed in Table 3.3-10; (4) add four instruments and delete seven instruments from Table 3.3-10; and (5) revise the TS Bases for the accident monitoring instrumentation. This is TVA's TS Change Request 89-30.

The modifications to upgrade the instruments to the provisions in Regulatory Guide (RG) 1.97, Revision 2, Instrumentation For Light-Water-Cooled Nuclear Power Plants To Assess Plant Conditions During And Following An Accident," December 1980, were completed during the Cycle 4 refueling outage for each unit (i.e., March to June, 1990, for Unit 1, and September to November, 1990, for Unit 2).

2.0 EVALUATION

The proposed changes in TS Change Request 89-30 can be broken down into five categories. These categories are listed above and they will be evaluated separately. These evaluations are given below:

2.1 Delete Table 4.3-7

Table 4.3-7 lists the surveillance requirements for the accident monitoring instrumentation given in Table 3.3-10. The surveillance requirements are monthly for a channel check and every refueling outage for the channel calibration. TVA has proposed to add these same requirements in SR 4.3.3.7 instead of Table 4.3-7, and then delete Table 4.3-7, because they are the same for every accident monitoring instrument listed in Table 3.3-10. Because the proposed changes do not change the surveillance requirements for the instruments in Table 4.3-7, the staff concludes that these changes are acceptable.

TVA also proposed to have the footnote for the containment area monitors in Table 4.3-7 to be a footnote to SR 4.3.3.7 with a reference to these monitors. This proposed change is an administrative change to continue the description of the channel calibration for these monitors with Table 4.3-7 deleted. The proposed change does not revise the description of the channel calibration for these monitors; therefore, the staff concludes that the proposed change is acceptable.

2.2 Transfer Action Statements to Table 3.3-10

TVA has proposed to have the action statements for TS 3/4.3.3.7 in Table 3.3-10. The intent is to transfer the requirements currently in the action statements from LCO 3.3.3.7 to Table 3.3-10. This format is used in other tables in the TSs. The proposed action statements in Table 3.3-10 were reviewed against the current ones to determine if any of the current requirements were being changed. This review took into account that the columns in Table 3.3-10 were being revised by the proposed changes. Instead of "Required Number of Channels" and "Minimum Channels Operable," TVA has proposed the following headings for the columns: "Total Number of Channels" and "Minimum Channels Required."

Based on the proposed minimum channels required in the proposed Table 3.3-10, all the instruments in the revised Table have proposed action statements which are at least the same as the current action statements and the same number of channels required to be operable except for the incore thermocouples. The following instruments have additional requirements in the proposed action statements which are beyond the current requirements: reactor coolant pressure, pressurizer level, and reactor coolant system subcooling margin monitor. The redundancy for the AFW flow is provided by the single AFW flow channel for each steam generator and the three AFW level control valve position indicators (i.e., a valve for each of the two motor-driven AFW flow paths and a valve for the turbine AFW flow path).

The incore thermocouples only have a requirement for the unit to go to hot shutdown if the number of operable channels is less than the minimum number required per core quadrant for 48 hours. They do not have a requirement that if the number of operable channels is one less than the total number of channels for seven days, the unit must go to hot shutdown. Because the minimum number of channels required is 1/core quadrant/train, or 8 per core, and there is a total of 65 incore thermocouple channels in the core, the staff concludes that the proposed changes for the incore thermocouples are acceptable.

For the subcooling margin monitor, the requirements in the current action statements for LCO 3.3.3.7 for an inoperable monitor were based on (1) the units each having only one monitor, (2) the monitors not meeting the requirements of TMI Action Plan Item II.F.2, and (3) the monitors not meeting provisions of RG 1.97. TVA committed to have the subcooling margin monitors meet the requirements of Item II.F.2 and provisions of RG 1.97 by the Cycle 4 refueling outage for both units in its letter dated August 14, 1985. This included increasing the number of monitors for each unit to two. This was accepted in the staff's Safety Evaluation dated November 22, 1985.

With the total number of these monitors being increased to two for each unit and these monitors meeting RG 1.97 and II.F.2 in the Cycle 4 refueling outage for both units, the current action statements for these monitors are not appropriate in that they do not require an eventual unit shutdown for an inoperable monitor(s) and instead rely on an additional shift crew member dedicated to and capable of determining the subcooling margin during an accident for indefinite unit operation with an inoperable monitor(s).

The action statements should be in accordance with the guidance in Generic Letter (GL) 83-37, "NUREG-0737 Technical Specifications," dated November 1, 1983. The proposed action statements for these monitors are not consistent with GL 83-37 and TVA has not provided justification for this inconsistency. Therefore, the staff will request TVA to propose the action statements in GL 83-37 or provide justification that the action statements in GL 83-37 do not apply to Sequoyah. Because the proposed action statements are consistent with the current action statements for these monitors, the staff concludes that the proposed action statements with the requirements for an additional shift crew member dedicated to determining the subcooling margin during an accident are acceptable until TVA has resolved this issue.

Therefore, based on the above, that all the instruments in Table 3.3-10 have proposed minimum channels required and action statements that meet the requirements in the current action statements or are acceptable, including the incore thermocouples and the subcooling margin monitors as discussed above, the staff concludes that the proposed changes for the total number of channels, minimum channels required, and action statements in Table 3.3-10 are acceptable. For the subcooling margin monitors, the staff will request TVA to address the action statements in GL 83-37.

For proposed Actions 1, 2, and 5, there is a note given with the action statements. The notes refer to all or one of the following tables in the TSs: Table 3.3-1, "Reactor Trip System Instrumentation," Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," and Table 3.3-9, "Remote Shutdown Monitoring Instrumentation." TVA is stating in these notes that there are applicable action requirements in these tables for some of the instrumentation listed in Table 3.3-10 and these action requirements should be referred to if the equipment is inoperable to determine which action requirements are the most restrictive. For example, because the source range neutron flux instrumentation is listed in both Tables 3.3-1 and 3.3-10, the action statements in both tables would apply if this instrumentation is inoperable. The more restrictive action statement for the situation the reactor was in would be the one that applied to the inoperable instrumentation at that time. This is additional information for the control room operators. The staff agrees that some of the instruments listed in Table 3.3-10 are also listed in Table 3.3-1, 3.3-3, and 3.3-9. Therefore, the notes provide useful information to the control room operators and the staff concludes that this proposed change is acceptable. It should be pointed out that the action statements for

Table 3.3-9 are, in fact, in LCO 3.3.3.5. The staff also concludes that proposed Actions 3, 4, and 6 do not need a similar note.

2.3 Add Information on Instrument Loops

TVA has proposed to add the numbers of the specific instrument loop for all instruments listed in Table 3.3-10, execpt the in-core thermocouples which have a total number of 65 channels. TVA provided the instrument loops for the following instruments that the staff did not agree could be deleted from Table 3.3-10 without further justification, as discussed in Section 2.4 below: wide range containment pressure and reactor vessel level indication system. This information is not used to decide if an instrument is operable or what action statement would apply if an instrument is inoperable. The numbers simply show the specific loops at Sequoyah for the accident monitoring instrumentation listed in the table. The number of instrument loops is equal to the total number of channels listed in Table 3.3-10 except for the reactor vessel level indication system which was three loops/channel. This change is administrative in nature. The staff concludes that the proposed change is acceptable.

2.4 Revise Instruments Listed in Table 3.3-10

TVA proposed to revise Table 3.3-10 to only list the Type A, Category 1 instruments of Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The Type A instrument variables in RG 1.97 provide primary information to the control room operators to permit them to take specified manually controlled actions for which no automatic action is provided and that are required for safety systems to accomplish their safety functions for design basis events. Category 1 instrument variables have design requirements for seismic and environmental qualification, application of single failure criteria, utilization of emergency standby power, control room display, continuous readout, and recording capability. TVA stated that the following instrumentation has been designated for Sequoyah as Type A, Category 1 instruments in accordance with the guidance given in RG 1.97 and a review of emergency procedures, functional restoration guidelines, and the Sequoyah Final Safety Analysis Report (FSAR), Chapter 15 (design basis accidents):

1. Reactor Coolant System (RCS) Hot Leg Water Temperature (Wide Range)

2. RCS Cold Leg Water Temperature (Wide Range)

3. Containment Pressure (Narrow Range)

4. Refueling Water Storage Tank Level

5. Reactor Coolant Pressure (Wide Range)

6. Pressurizer Level

7. Steam Line Pressure

8. Steam Generator (SG) Level (Wide Range)

9. SG Level (Narrow Range)

10. Auxiliary Feedwater (AFW) Flow Rate and Level Control Valve Position

11. RCS Subcooling Margin Monitor

12. Containment Sump Water Level (Wide Range)

13. Incore Thermocouples

14. Containment Area Radiation Monitors (Upper and Lower)

15. Containment Hydrogen Monitors

16. Neutron Flux Monitors (Source and Intermediate Range)

The proposed TS change incorporates the above instruments into the accident monitoring instrumentation Table 3.3-10 except for the containment hydrogen monitors which are listed in TS 3/4.6.4, Combustible Gas Control. Based on the existing instruments in Table 3.3-10, the proposed TS change means that (1) the position indication for the pressurizer power-operated relief and block valves, and the safety valves; the wide-range containment pressure instrumentation; the reactor vessel level instrumentation system; and the high-range and mid-range noble gas effluent monitors are being deleted from Table 3.3-10 and (2) the narrow-range containment pressure indication, the auxiliary feedwater level control valve position indication, and the source range and intermediate-range neutron flux instruments are being added to Table 3.3-10. There are also footnotes in Table 3.3-10, which are associated with the instruments being proposed to be deleted, that TVA has also proposed to delete.

TVA provided the following justifications to support the deletion of the six instruments from Table 3.3-10:

2.4.1 Pressurizer Power-Operated Relief Valve (PORV) Position Indication

PORV position indication is used by the operator to ensure that the reactor coolant system (RCS) pressure boundary is intact when RCS pressure is decreasing. However, RCS pressure has been included as a Type A variable and is considered to provide the primary information to the operator. If RCS pressure is decreasing, the operator can take conservative actions by attempting to close the pressurizer PORV or block valve even if the position indication is not available. Thus, the pressurizer PORV position does not meet the criteria of a Type A variable. The PORV position indication for Sequoyah is designated as a Type D, Category 2 variable which is consistent with the guidance in RG 1.97. In addition, the footnote in Table 3.3-10 associated with PORV position indication is no longer applicable.

2.4.2 Pressurizer PORV Block Valve Position Indication

The PORV block valve position indication provides a backup to the PORV for ensuring the RCS pressure boundary is intact when RCS pressure is decreasing. RCS pressure has been included as a Type A variable and is considered to provide the primary information to the operator. If RCS pressure is decreasing, the operator can take conservative actions by attempting to close the pressurizer PORV or block valve even if the position indication is not available. Thus, the block valve position does not meet the criteria of a Type A variable. The PORV block valve position indication for Sequoyah is designated as a Type D, Category 2 variable which is consistent with the guidance in RG 1.97. In addition, the footnote in Table 3.3-10 associated with PORV block valve indication is no longer applicable.

2.4.3 Pressurizer Safety Valve Position Indication

Safety valve position indication can be used by the operator to determine if the RCS pressure boundary is intact when RCS pressure is decreasing; however, these valves are just three of several RCS valves that could be open and breaching the pressure boundary. RCS pressure has been included as a Type A variable and is considered to provide the primary information to the operator that RCS pressure is decreasing. Thus, safety valve position indication does not meet the criteria of a Type A variable. The safety valve position indication for Sequoyah is designated as a Type D, Category 2 variable which is consistent with the guidance in RG 1.97.

2.4.4 Wide Range Containment Pressure

TVA is proposing to replace the wide range containment pressure instruments in Table 3.3-10 by the narrow range containment pressure instruments. These instruments are direct indication of containment pressure but the wide range containment pressure is designated for Sequoyah as Type B, Category 1 in accordance with the guidance in RG 1.97 and the narrow-range containment pressure is designated for Sequoyah as Type A, Category 1.

2.4.5 Reactor Vessel Level Indication System (RVLIS)

RVLIS is considered to be a direct indication of core cooling and is designated for Sequoyah as Type B, Category 1 in accordance with the guidance in RG 1.97, Table 2. In addition, the footnote associated with RVLIS is no longer applicable.

2.4.6 Shield Building Exhaust Vent (High-Range/Mid-Range Noble Gas)

The shield building exhaust vent serves as a nonisolable primary release point for detection of airborne radioactive material following an accident. In accordance with the guidance in RG 1.97, the noble gas monitors for Sequoyah have been designated as Category 2. In addition to being Type E variable, TVA also designated this vent (noble gas) as a Type C variable because it provides information that would indicate a breach in containment integrity. The high-range/mid-range noble gas monitors for the shield building exhaust vent do not meet the criteria of a Type A variable.

2.4.7 Condenser Vacuum Exhaust Vent (High-Range/Mid-Range Noble Gas)

The function of the condenser vacuum exhaust during an accident is the detection of secondary-side radioactivity from the primary RCS side. Main steam line radiation levels, steam generator (SG) blowdown radiation levels, and SG sampling provide similar information for detecting secondary-side radiation. The Sequoyah emergency instructions direct the operator to use secondary-side radiation levels to diagnose a SG tube rupture event. Identification of this event and the subsequent required manual actions are taken based on increasing SG level with no AFW flow. The SG level (narrow-range) and AFW flow are both classified as Type A, Category 1 variables. Consequently, the condenser vacuum exhaust vent (high-range/mid-range) noble gas monitors have been designated as a Type E, Category 2 variable for Sequoyah. In addition to being Type E, TVA also designated the condenser vacuum exhaust (noble gas) as a Type C variable because it provides information that would indicate a breach in the RCS pressure boundary.

2.4.8 RG 1.97 Type A, Category 1 Instrumentation

For the instrumentation listed above which TVA proposed to be deleted from Table 3.3-10, the instruments are not associated with RG 1.97 Type A, Category 1 variables for Sequoyah. The RG 1.97 type and category of the instruments for Sequoyah are listed in Attachment 1 to TVA's submittal on RG 1.97, dated May 7, 1990. The entire list of RG 1.97 Type A, Category 1 variables for Sequoyah are associated with the 16 instruments listed above. Therefore, all of the RG 1.97 Type A, Category 1 variable instrumentation are being proposed for Table 3.3-10 with the exception of the containment hydrogen monitors which are given in TS 3/4.6.4.1.

It is the staff's position that Table 3.3-10 should list all RG 1.97 Category 1 instrumentation and not just the Type A, Category 1 instrumentation. Category 1 applies to the key safety instrumentation for the units. It does not apply to instrumentation designated as indicating the status of operating systems (i.e., Category 2) or as backup and diagnostic (i.e., Category 3). Therefore, because of their implied safety significance, all the Category 1 instrumentation should be listed in Table 3.3-10 of the TSs unless the licensee can justify that the safety significance of the instrumentation is not sufficient to warrant including the instrumentation in the TSs.

In its application, TVA has not proposed to have the following RG 1.97 Category 1 instrumentation in the TSs: wide-range containment pressure instrumentation, reactor vessel level instrumentation, containment isolation valve position, and the essential raw cooling water system to auxiliary feedwater pump valve position indication. These four instruments are identified as RG 1.97 Category 1 instrumentation for Sequoyah in the TVA response to RG 1.97 dated May 7, 1990. The first two instruments are currently in Table 3.3-10 and TVA has proposed to delete them because they are not RG 1.97 Type A, Category 1 instrumentation for Sequoyah. The latter two instruments are not currently in Table 3.3-10 and TVA has not proposed to add them to the TSs because they are also not RG 1.97 Type A, Category 1 instrumentation for Sequoyah.

For these four instruments, TVA has not provided justification that the safety significance of these instruments is not sufficient to warrant having them included in the TSs. TVA's justification for deleting the wide-range containment pressure and reactor vessel level instrumentation was only that they were not RG 1.97 Type A, Category 1 instruments for Sequoyah. This justification is not sufficient for the staff to conclude that the proposed change to delete these instruments is acceptable.

The staff does not agree that the wide-range containment pressure and reactor vessel level instrumentation should be deleted from Table 3.3-10. However, the other changes to revise Table 3.3-10 by (1) deleting the pressurizer PORV and block valve position indication, pressurizer safety valve position indication, shield building exhaust vent high-range and mid-range noble gas monitor, and condenser vacuum exhaust high-range and mid-range noble gas monitor, and (2) adding the source-range and intermediate-range neutron flux monitors and the narrow-range containment pressure instrumentation are acceptable. These changes are acceptable because TVA is proposing to delete instrumentation which are not RG 1.97 Category 1 and add instrumentation which are RG 1.97 Category 1, for Sequoyah.

The staff will request TVA to (1) withdraw its request to delete the wide-range containment pressure and reactor vessel level instrumentation and (2) add the containment isolation valve position indication and the essential raw cooling water system to auxiliary feedwater pump valve position indication.

The proposed Table 3.3-10 is in a different format than the current Table 3.3-10. Therefore, in keeping the wide-range containment pressure and reactor vessel level instrumentation, as discussed above, in Table 3.3-10 means that the format of the data and action statements for these instruments must be revised. This was discussed with TVA and agreed upon during the visit of the NRC Sequoyah Project Manager to the Sequoyah site on August 16, 1990. The format of the data and action statements for the wide range containment pressure and reactor vessel level instrumentation were revised from that in TS 3/4.3.3.7 of the current TSs but the requirements on the instruments in the TSs are not being changed. Therefore, the revision to the format of the data and action statements for these instruments in TS 3/4.3.3.7 does not change the substance of the proposed action in the Federal Register Notice (55 FR 6118) which was published on February 21, 1990 for the proposed amendments and does not affect the staff's initial determination of no significant hazards consideration in that notice.

The footnotes associated with the instruments being proposed to be deleted from Table 3.3-10, including the reactor vessel level instruments, are also proposed by TVA to be deleted because they are no longer applicable. The staff agrees with this conclusion and concludes that the proposed changes to delete the footnotes are acceptable.

2.5 TS Bases Change

TVA stated that the bases for TS 3/4.3.3.7, Accident Monitoring Instrumentation, have been revised to do the following: (1) provide TVA's licensing position for the inclusion of RG 1.97 Type A, Category 1 instrumentation in the TSs; (2) update the reference to RG 1.97 to reflect TVA's commitments for Revision 2 of RG 1.97, dated December 1990; (3) delete portions of the bases involving the use of acoustic monitors are one of the two required channels for determining pressurizer safety valve position; (4) describe the redundant channel capability for AFW flow instrumentation and AFW valve position indication; and (5) reference TS 3.6.4.1 to specify operability requirements for containment hydrogen monitors. The TS bases for TS 3/4.6.4, Combustible Gas Control, have also been modified to reference the containment hydrogen monitors as RG 1.97 accident monitoring instrumentation.

TVA proposed entirely new text for the bases of TS 3/4.3.3.7. The text is considered accurate and correct and consistent with the above accepted TS changes, except for TVA's licensing position that only RG 1.97 Type A, Category 1 instrumentation should be in the TSs. It is the staff's position that all RG 1.97 Category 1 instrumentation should be listed in the TSs unless the licensee can justify that the safety significance of the specific instrumentation is not sufficient to warrant having the instrumentation in the TSs. Therefore, the staff concludes that the proposed changes to the basis of TS 3/4.3.3.7 are acceptable except for the statements concerning TVA's position on RG 1.97 Type A, Category 1 instrumentation. TVA will be requested to withdraw or revise this proposed change to the basis of TS 3/4.3.3.7.

TVA also proposed to state in the bases for TS Section 3/4.6.4, Combustible Gas Control, that the hydrogen monitors of TS 3.6.4.1 are part of the accident monitoring instrumentation that are RG 1.97 Type A, Category 1 variables for Sequoyah. This is correct and, therefore, the staff concludes that this proposed change is acceptable.

2.6 Modifications To Meet RG 1.97

TVA stated that the instrument modifications to meet RG 1.97 were scheduled for the Cycle 4 refueling outage on each unit. Unit 1 is the lead unit with the Unit 1 Cycle 4 refueling outage completed in May 1990 and the Unit 2 Cycle 4 refueling outage having been completed in November 1990. The majority of the RG 1.97 modifications involved instrumentation that can only be installed or modified during a plant shutdown. TVA submitted the proposed TS change to coincide with the Unit 1 modification schedule which was completed by the Unit 1 restart in May 1990 from its Cycle 4 refueling outage.

2.7 Evaluation of RG 1.97

TVA has submitted its response to RG 1.97 for Sequoyah in letters dated December 28, 1988, September 14, 1989, and May 7, 1990. This is being reviewed by the staff and will be the subject of a separate evaluation. The staff is in agreement with TVA's submittal dated May 7, 1990 as to the RG 1.97 Category 1 instrumentation for Sequoyah.

2.8 Conclusion

Based on Sections 2.1 to 2.5 above, the staff concludes that the proposed changes in TVA's TS Change Request 89-30 are acceptable except for the proposed changes discussed in Sections 2.4.8 and 2.5 above. The staff concludes that all RG 1.97 Category 1 instrumentation should be listed in the TSs. Therefore, the staff will request TVA to (1) withdraw its request to delete the wide-range containment pressure instrumentation and the reactor vessel instrumentation from Table 3.3-10, (2) propose the addition of the containment isolation valve position indication and the essential raw cooling water system to auxiliary feedwater pump valve position indication to Table 3.3-10, and (3) withdraw or revise its request to have the basis for TS 3/4.3.3.7 be only the RG 1.97 Type A, Category 1 instrumentation.

The staff concludes in Section 2.2 that the action statements for inoperable subcooling margin monitors in Table 3.3-10 should be consistent with the guidance in GL 83-37. Therefore, the staff will request TVA to propose appropriate action statements for these monitors consistent with the GL or provide a justification that the action statements do not apply to Sequoyah.

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 be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 6118) on February 21, 1990 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 (3) the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

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