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December 17, 1993
Refer to: RC-93-0310

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. G. F. Wunder

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO NPF-12
TECHNICAL SPECIFICATION CHANGE - TSP 930011
ACCIDENT MONITORING INSTRUMENTATION

In accordance with 10CFR50.90, South Carolina Electric & Gas Company (SCE&G) is submitting an amendment request to License NPF-12 to change the Technical Specifications (TS) for VCSNS. The proposed change is a general revision to the Accident Monitoring Instrumentation, TS 3/4.3.3.6, and associated Bases 3/4.3.3.6, and relocates the Hydrogen Monitors, TS 3/4.6.5.1, and the Reactor Building Area High Range Radiation Monitors, TS 3/4.3.3.1, Tables 3.3-6 and 4.3-3, item 1.c, into the Accident Monitoring TS. Accordingly, SCE&G withdraws the request made in the application dated July 22, 1992 (TSP 890005), to revise the Accident Monitoring Instrumentation.

The proposed change provides requirements for key plant variables consistent with VCSNS's accepted program on Revision 3, 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 proposed TS change is consistent with NUREG-1431, Standard Technical Specifications for Westinghouse Plants, except for the Condensate Storage Tank (CST) Level and the Containment Isolation Valve Position Indication. VCSNS CST Level is neither a Category 1 variable nor a key variable, based on the fact that the operator does not take any prescribed manual action based on CST level indication. Normally, the Emergency Feedwater (EFW) System is aligned to take suction from the CST; once its inventory is depleted, the EFW pump suction will automatically switch to the Service Water System. Therefore, the CST Level does not provide primary information to the operator and does not meet the criteria of a key variable. For the Containment Isolation Valve (CIV) Position Indication, redundant isolation valves are provided for lines penetrating the containment, and only active valves have position indication. Check valves have no position indication and are specifically excluded by the Regulatory Guide.

A001

The amendment request is contained in the following enclosures:

Enclosure 1	References
Enclosure 2	Proposed TS Change
Enclosure 3	Description and Safety Evaluation
Enclosure 4	Determination of No Significant Hazards Consideration

The proposed TS change has been reviewed and approved by the Plant Safety Review Committee and the Nuclear Safety Review Committee.

I declare that the statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Should you have any questions concerning this issue, please call Mr. Manuel W. Gutierrez at (803) 345-4392 at your convenience.

Very truly yours,

John L. Skolds

John L. Skolds

MWG:lcd
Enclosures

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Central File System
File (813.20)

REFERENCES

1. South Carolina Electric & Gas Co. (SCE&G) letter to NRC dated April 15, 1985 - "V. C. Summer Nuclear Station (VCSNS) - Generic Letter 82-33, Emergency Response Capability Supplement 1 to NUREG-0737"

2. NRC letter to SCE&G dated June 18, 1986 - "Regulatory Guide 1.97 Request for Additional Information"

3. SCE&G letter to NRC dated August 18, 1986 - VCSNS - "Regulatory Guide 1.97 Request for Additional Information"

4. SCE&G letter to NRC dated October 31, 1986 - VCSNS - "Regulatory Guide 1.97 Request for Additional Information"

5. NRC letter to SCE&G dated November 13, 1987 - "Regulatory Guide 1.97 (TAC No.51137)"

6. SCE&G letter to NRC dated February 16, 1988 - VCSNS - "Regulatory Guide 1.97 Open Items"

7. SCE&G letter to NRC dated May 20, 1988 - VCSNS - "Generic Letter 82-33, Regulatory Guide 1.97"

8. NRC letter to SCE&G dated July 27, 1988 - "Safety Evaluation on Conformance to Regulatory Guide 1.97 (TAC No.51137)"

9. SCE&G letter to NRC dated September 4, 1992 - "Instrumentation to Follow the Course of an Accident, Regulatory Guide 1.97"

10. SCE&G letter to NRC dated September 14, 1992 - "Technical Specification Change - TSP 890005, Accident Monitoring Instrumentation"

Enclosure 2 to Document Control Desk Letter
TSP 930011
Page 1

PROPOSED TECHNICAL SPECIFICATION CHANGE - TSP 930011
VIRGIL C. SUMMER NUCLEAR STATION

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.6 CONTAINMENT SYSTEMS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	≤ 1 R/hr	1 - 10^{5} mr/hr	28
DELETE ~~c. Reactor Building Area i. High Range RM-G7 and High Range RM-G18~~	~~2~~	~~1, 2, 3 & 4~~	~~N/A~~	~~10 - 10^{7} R/hr 1 - 10^{7} R/hr~~	~~30~~
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	< 1 x 10^{-5} μCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	≤ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	S	R	M	6
DELETE ~~c. Reactor Building Area~~				
~~i. High Range (RM-G7)~~	~~S~~	~~R***~~	~~M~~	~~1, 2, 3 & 4~~
~~ii. High Range (RM-G18)~~	~~S~~	~~R***~~	~~M~~	~~1, 2, 3 & 4~~
2. PROCESS MONITORS				
a. Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	S	R	M	6
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

*With fuel in the storage pool or building

**With irradiated fuel in the storage pool

DELETE ~~***Channel Calibration will consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source~~

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channels to OPERABLE status witin 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.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

REPLACE WITH NEW PAGES

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Building Wide Range Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/stm. gen.	1/steam generator
7. Steam Generator Water Level - Wide Range	1/stm. gen.	1/steam generator
8. Emergency Feedwater Flow	1/stm. gen.	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Boric Acid Tank Water Level	2/tank	1/tank
11. Reactor Building Spray Pump Discharge Flow	2	1
12. Reactor Building Temperature	2	1

REPLACE WITH NEW TABLE 3.3-10

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL OF CHANNELS	MINIMUM CHANNELS OPERABLE
13. Reactor Building/RHR Sump Level	2	1
14. DELETED.		
15. Condensate Storage Tank Level	2	1
16. Reactor Building Cooling Unit Service Water Flow	2	1
17. Service Water Temperature-Reactor Building Cooling Unit (Inlet and Discharge)	2 pairs	1 pair
18. NaOH Storage Tank Level	2	1
19. Reactor Coolant System Subcooling Margin Monitor	2	1
20. Pressurizer PORV Position Indicator	2/valve*	1/valve*
21. Pressurizer PORV Block Valve Position Indicator	1/valve	1/valve
22. Pressurizer Safety Valve Position Indicator	2/valve	1/valve
23. In-Core Thermocouples	4/core quadrant	2/core quadrant
24. Reactor Vessel Level	2	1

*Not required when the associated block valve is closed per Specification 3.4.4.

REPLACE WITH NEW TABLE 3.3-10

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Building Wide Range Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Wide Range	M	R
8. Emergency Feedwater Flow	M	R
9. RWST Water Level	M	R
10. Boric Acid Tank Solution Level	M	R
11. Reactor Building Spray Pump Discharge Flow	M	R

DELETED

TABLE 4.3-7 (continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
12. Reactor Building Temperature	M	R
13. Reactor Building/RHR Sump Level	M	R
14. DELETED.		
15. Condensate Storage Tank Level	M	R
16. Reactor Building Cooling Unit Service Water Flow	M	R
17. Service Water Temperature - Reactor Building Cooling Unit (Inlet and Discharge)	M	R
18. NaOH Storage Tank Level	M	R
19. Reactor Coolant System Subcooling Margin Monitor	M	R
20. Pressurizer PORV Position Indicator	M*	R*
21. Pressurizer PORV Block Valve Position Indicator	M	R
22. Pressurizer Safety Valve Position Indicator	M	R
23. In-Core Thermocouples	M	R
24. Reactor Vessel Level	M	R

DELETED

*Not required when the associated block valve is closed per Specification 3.4.4.

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

DELETED

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing 19.8 ±0.1 volume percent hydrogen, balance nitrogen.

INSTRUMENTATION

BASES

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. 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 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 Criteria 19 of 10 CFR 50.

~~3/4.3.3.6 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, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."~~

REPLACE

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	≤ 1 R/hr	1 - 10^{5} mR/hr	28
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	≤ 1 x 10^{-5} μCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	≤ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	S	R	M	6
2. PROCESS MONITORS				
a. Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	S	R	M	6
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 With the number of Hydrogen monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, restore at least one monitor to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Reactor Building Pressure - Narrow Range Instrument Loop/Indicator: Channel D IPT-951/IPI-951 Channel B IPT-952/IPI-952	2	1
2.	Reactor Building Pressure - Wide Range Instrument Loop/Indicator: Channel D IPT-954A/IPI-954A Channel E IPT-954B/IPI-954B	2	1
3.	Reactor Building Radiation Level - High Range Instrument Loop/Indicator: Channel A RMG-18 Channel B RMG-7	2	1
4.	Reactor Building Hydrogen Concentration Instrument Loop/Indicator: Channel A IAE-8263A/ICI-8257 Channel B IAE-8263B/ICI-8258	2	1
5.	Reactor Building/RHR Sump Level Instrument Loop/Indicator: Channel A ILT-1969/ILI-1969 Channel B ILT-1970/ILI-1970	2	1
6.	Reactor Coolant Outlet Temperature - T_{Hot} - Wide Range Instrument Loop/Indicator: Channel A ITE-413/ITI-413 Channel A ITE-423/ITI-423 Channel E ITE-433/ITR-413	2	1
7.	Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range Instrument/Loop Indicator: Channel E ITE-410/ITI-410 Channel E ITE-420/ITI-420 Channel E ITE-430/ITR-410	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT		REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
8.	Reactor Coolant Pressure - Wide Range Instrument Loop/Indicator: Channel E IPT-402/IPI-402 Channel A IPT-403/IPI-403		2	1
9.	Pressurizer Water Level Instrument Loop/Indicator: Channel A ILT-459/ILI-459A Channel D ILT-460/ILI-460 Channel B ILT-461/ILI-461		2	1
10.	Reactor Coolant System Subcooling Margin Instrument Loop/Indicator: Channel A ITM-499A Channel B ITM-499B		2	1
11.	Reactor Vessel Level Instrument Loop/Indicator: Channel A ILT-1311/ILI-1311, ILT-1312/ILI-1312 Channel B ILT-1321/ILI-1321, ILT-1322/ILI-1322		2	1
12.	Core Exit Temperature Instrument Loop/Indicator: Channel A ITEs 2, 4, 9, 12, 13, 15, 19, 21, 22, 23, 24, 25, 26, 27, 28, 29, 31, 32, 33, 35, 39, 41, 42, 45, 46, 47 (Primary display is the plant computer) (Backup displays are ITM 499 A&B)	Channel B ITEs 1, 3, 5, 6, 7, 8, 10, 11, 14, 16, 17, 18, 20, 30, 34, 36, 37, 38, 40, 43, 44, 48, 49, 50, 51	4/core quadrant/ channel	2/core quadrant/ channel
13.	Neutron Flux Instrument Loop/Indicator: Channel 1 INI-35 Channel 2 INI-36		2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Steam Line Pressure Instrument Loop/Indicator: SG A IPTs-474, 475, 476/IPIs-474, 475, 476 SG B IPTs-484, 485, 486/IPIs-484, 485, 486 SG C IPTs-494, 495, 496/IPIs-494, 495, 496	2/stm. gen.	1/stm. gen.
15. Steam Generator Water Level - Wide Range Instrument Loop/Indicator: SG A ILT-477/ILI-477 SG B ILT-487/ILI-487 SG C ILT-497/ILI-497	1/stm. gen.	1/stm. gen.
16. Steam Generator Water Level - Narrow Range Instrument Loop/Indicator: SG A ILTs 474, 475, 476/ILIs 474, 475, 476 SG B ILTs 484, 485, 486/ILIs 484, 485, 486 SG C ILTs 494, 495, 496/ILIs 494, 495, 496	2/stm. gen.	1/stm. gen.
17. Emergency Feedwater Flow Instrument Loop/Indicator: Channel A SG A IFT-3561/IFI-3561 SG B IFT-3571/IFI-3571 SG C IFT-3581/IFI-3581 Channel B SG A IFT-3561A/IFI-3561B SG B IFT-3571A/IFI-3571B SG C IFT-3581A/IFI-3581B	2/stm. gen.	1/stm. gen.
18. Refueling Water Storage Tank Level Instrument Loop/Indicator: Channel A ILT-990/ILI-990 Channel B ILT-992/ILI-992	2	1

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3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. 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 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 Criteria 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The PAM Instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required to perform certain manual actions specified in the Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.3.6 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

Type A and Category I variables are required to meet Regulatory Guide 1.97 Category I design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

The following are discussions of specified instrument functions listed in Table 3.3-10.

1. & 2. Reactor Building Pressure

Reactor Building Pressure is provided for verification of RCS and containment OPERABILITY. Reactor Building Pressure is used to verify closure of main steam isolation valves (MSIVs), and containment spray Phase B isolation. Other manual actions based on Reactor Building Pressure include: stopping the RCPs, stopping containment spray pumps, and starting RHR pumps. Reactor Building Pressure indications are also required to calculate reactor vessel vent times.

3. Reactor Building Radiation Level

Reactor Building Radiation Level is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Reactor Building Radiation Level is used to determine if a high energy line break (HELB) has occurred and whether the event is inside or outside of containment.

4. Reactor Building Hydrogen Concentration

Reactor Building Hydrogen Concentration Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions. Reactor Building Hydrogen concentration is used by the operator to calculate reactor vessel vent time and is monitored as a criterion for continuing vessel venting when attempting to collapse voids.

5. Reactor Building/RHR Sump Water Level (Wide Range)

Reactor Building/RHR Sump Water Level is provided for verification and long term surveillance of RCS integrity. Reactor Building/RHR Sump Water Level is used to determine: containment sump level accident diagnosis; when to begin the recirculation procedure; and whether to terminate SI, if still in progress.

6. & 7. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures variables provide verification of core cooling and long term RCS surveillance. In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

8. Reactor Coolant Pressure

RCS pressure provides verification of core cooling and RCS integrity long term surveillance. RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs). RCS pressure can be used: to determine whether to terminate actuated SI or to reinitiate stopped SI; to determine when to reset SI and shut off low head SI; to manually restart low head SI; as reactor coolant pump (RCP) trip criteria; and to make a determination on the nature of the accident in progress and where to go next in the procedure. RCS pressure is also related to three decisions about depressurization. They are: to determine whether to proceed with primary system depressurization; to verify termination of depressurization; and to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization. A final use of RCS pressure is to determine whether to operate the pressurizer heaters. RCS pressure is used by the operator to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

9. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

10. Reactor Coolant System Subcooling Margin

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control.

11. Reactor Vessel Level

Reactor Vessel Level is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

Reactor Vessel Level Monitoring provides direct measurement of collapsed liquid level above the fuel alignment plate. Collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of collapsed water level is selected because it is a direct indication of the water inventory.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

12. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature, in each quadrant, provide indication of radial distribution of coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient.

13. Neutron Flux

Neutron Flux indication is provided to verify reactor shutdown. Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

14. Steam Line Pressure

Steam Line Pressure is used to: identify and isolate a faulted steam generator; maintain an adequate reactor heat sink; verify that EFW to steam generator associated with pipe rupture is isolated; and monitor secondary side steam pressure to: verify operation of pressure control steam dump system, monitor RCS cooldown rate, and maintain plant in cold shutdown condition.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

15. & 16. Steam Generator Water Level

SG Water Level is provided to monitor operation of decay heat removal via the SGs. Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater Control System.

SG Water Level is used to: identify the faulted SG following a tube rupture; verify that the intact SGs are an adequate heat sink for the reactor; determine the nature of the accident in progress (e.g., verify an SGTR); and verify unit conditions for termination of SI during secondary unit HELBs outside containment. Operator action is based on control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, SG-condenser mode of heat transfer is necessary to remove decay heat. Operator action is required to manually raise and control SG level to establish heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated level reaches its setpoint.

17. Emergency Feedwater Flow

EFW Flow is provided to monitor operation of decay heat removal via the SGs. EFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 gpm to 1000 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel. EFW flow is used three ways: to verify delivery of EFW flow to the SGs; to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and to regulate EFW flow so that the SG tubes remain covered. Operator action is required to throttle EFW flow during an SLB accident to prevent the EFW pumps from operating in runout conditions. EFW flow is also used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the Primary indication used by the operator to ensure an adequate inventory is SG level.

18. Refueling Water Storage Tank (RWST) Level

RWST Level is provided to ensure water supply for ECCS. RWST low level indications are used by the operator as the basis for aligning ECCS suction to the containment sump and stopping all pumps taking suction from the RWST on low-low level.

PROPOSED TECHNICAL SPECIFICATION CHANGE - TSP 930011
VIRGIL C. SUMMER NUCLEAR STATION

DESCRIPTION AND SAFETY EVALUATION

DESCRIPTION OF AMENDMENT REQUEST

SCE&G proposes to modify the VCSNS TS to revise TS 3/4.3.3.6, "Accident Monitoring Instrumentation," and associated Bases 3/4.3.3.6. This change will reflect the plant variables (key variables) that provide primary information required to permit control room operators to take specified manually controlled actions (for which no automatic control is provided) after initial stages of an accident and that are required for safety systems to accomplish their safety function for design basis accident events.

The instrumentation, to be included in Table 3.3-10, is designated type A category 1, except for the Subcooling Margin Monitor which is category 2 but meets the intent of category 1, and non-type A category 1. This designation is in accordance with the guidance given in Regulatory Guide (RG) 1.97, revision 3, and SCE&G submittal of April 15, 1985, found acceptable by the NRC in its Safety Evaluation Reports dated November 13, 1987, and July 27, 1988. The proposed TS change is consistent with NUREG-1431, Standard Technical Specifications for Westinghouse Plants. The following instrumentation shall be included in Table 3.3-10:

1. Reactor Building Pressure - Narrow Range
2. Reactor Building Pressure - Wide Range
3. Reactor Building Radiation Level - High Range
4. Reactor Building H_2 Concentration
5. Reactor Building/RHR Sump Level
6. Reactor Coolant Outlet Temperature -T_{Hot} - Wide Range
7. Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range
8. Reactor Coolant Pressure - Wide Range
9. Pressurizer Water Level
10. Reactor Coolant System Subcooling Margin
11. Reactor Vessel Water Level
12. Core Exit Temperature
13. Neutron Flux
14. Steam Line Pressure
15. Steam Generator Water Level - Wide Range
16. Steam Generator Water Level - Narrow Range
17. Emergency Feedwater Flow
18. Refueling Water Storage Tank Level

The following non-Category 1 instrumentation shall be deleted from Table 3.3-10:

1. Boric Acid Tank Water Level
2. Reactor Building Spray Pump Discharge Flow
3. Reactor Building Temperature
4. Condensate Storage Tank Level
5. Reactor Building Cooling Unit Service Water Flow
6. Service Water Temperature-Reactor Building Cooling Unit (Inlet and Discharge)
7. Pressurizer PORV Position Indicator
8. Pressurizer PORV Block Valve Position Indicator
9. Pressurizer Safety Valve Position Indicator
10. Sodium Hydroxide Storage Tank Level

Table 4.3-7 shall be deleted.

The TS for the Reactor Building Area High Range Radiation Monitors, TS 3/4.3.3.1, Table 3.3-6 and 4.3-3, Item 1.c, and the Hydrogen Monitors, TS 3/4.6.5.1, have been relocated into this TS in order to have a consistent set of technical specifications.

The allowable outage time for the accident monitoring instruments has been changed from 7 to 30 days (for conditions in which the number of OPERABLE accident monitoring channels is less than the Required Number of Channels) and from 48 hours to 7 days (for conditions in which the number of OPERABLE accident monitoring channels is less than the Minimum Channels Operable), requiring a reduction to mode 3 (HOT STANDBY) within 6 hours and then to mode 4 (HOT SHUTDOWN) within the next 12 hours.

The mode APPLICABILITY for the Reactor Building Hydrogen Monitors was changed to include mode 3; the Reactor Building Area High Range Radiation Monitor was changed to delete mode 4.

Administrative changes were made to TS 3/4.3.3.6, Table 3.3-10.

SAFETY EVALUATION

The proposed TS change incorporates the type A category 1 and non-type A category 1 variables into VCSNS's "Accident Monitoring Instrumentation, Table 3.3-10." VCSNS's selection of category 1 key variables is documented in its Summary Report on Regulatory Guide 1.97, originally submitted to the NRC on April 15, 1985, amended by various letters referenced in this amendment request (Enclosure 1), and evaluated and accepted by the NRC in its letters of November 13, 1987, and July 27, 1988, and attendant Safety Evaluations.

Technical specifications for the Hydrogen Monitors, TS 3/4.6.5.1, and the Reactor Building Area High Range Radiation Monitors, TS 3/4.3.3.1, Tables 3.3-6 and 4.3-3, are relocated into the Accident Monitoring TS in order to provide a coherent presentation of specifications covering the same subject matter, thus, promoting better understanding and use of technical specifications by plant personnel.

The increase of the allowable outage time from 7 to 30 days in action statement a, applicable to the condition in which the number of OPERABLE accident monitoring channels is less than the Required Number of Channels, is based on: VCSNS's low failure rates for these types of components, the availability of a remaining operable channel, the passive nature of the channel, and the low probability of an event requiring accident monitoring instrumentation during this interval.

The increase of the allowable outage time from 48 hours to 7 days in action statement b.3, applicable to the condition in which the number of OPERABLE accident monitoring channels is less than the Minimum Channels Operable, is based on the relatively low probability of an event requiring accident monitoring instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a function is not acceptable because the alternate indications may not fully meet the requirements applied to the accident monitoring instrumentation. Therefore, requiring restoration of one inoperable channel of the function limits the risk that the accident monitoring function will be degraded should an accident occur.

The change of action statement b.3 requiring the plant to go to HOT STANDBY and then to HOT SHUTDOWN has no impact on the safety of the plant since it merely identifies the proper stages of plant shutdown which assumes that the plant is at 0% power.

PROPOSED TECHNICAL SPECIFICATION CHANGE - TSP 930011
VIRGIL C. SUMMER NUCLEAR STATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF AMENDMENT REQUEST

SCE&G proposes to modify the VCSNS TS to revise TS 3/4.3.3.6, "Accident Monitoring Instrumentation," and associated Bases 3/4.3.3.6. This change will reflect the plant variables (key variables) that provide primary information required to permit control room operators to take specified manually controlled actions (for which no automatic control is provided) after initial stages of an accident and that are required for safety systems to accomplish their safety function for design basis accident events.

The instrumentation, to be included in Table 3.3-10, is designated type A category 1, except for the Subcooling Margin Monitor which is category 2 but meets the intent of category 1, and non-type A category 1. This designation is in accordance with the guidance given in Regulatory Guide (RG) 1.97, revision 3, and SCE&G submittal of April 15, 1985 found acceptable by the NRC in its Safety Evaluation Reports dated November 13, 1987, and July 27, 1988. The proposed TS change is consistent with NUREG-1431, <u>Standard Technical Specifications for Westinghouse Plants</u>. The following instrumentation shall be included in Table 3.3-10:

1. Reactor Building Pressure - Narrow Range
2. Reactor Building Pressure - Wide Range
3. Reactor Building Radiation Level - High Range
4. Reactor Building H_2 Concentration
5. Reactor Building/RHR Sump Level
6. Reactor Coolant Outlet Temperature -T_{Hot} - Wide Range
7. Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range
8. Reactor Coolant Pressure - Wide Range
9. Pressurizer Water Level
10. Reactor Coolant System Subcooling Margin
11. Reactor Vessel Water Level
12. Core Exit Temperature
13. Neutron Flux
14. Steam Line Pressure
15. Steam Generator Water Level - Wide Range
16. Steam Generator Water Level - Narrow Range
17. Emergency Feedwater Flow
18. Refueling Water Storage Tank Level

The following non-Category 1 instrumentation shall be deleted from Table 3.3-10:

1. Boric Acid Tank Water Level
2. Reactor Building Spray Pump Discharge Flow
3. Reactor Building Temperature
4. Condensate Storage Tank Level
5. Reactor Building Cooling Unit Service Water Flow
6. Service Water Temperature-Reactor Building Cooling Unit (Inlet and Discharge)
7. Pressurizer PORV Position Indicator
8. Pressurizer PORV Block Valve Position Indicator
9. Pressurizer Safety Valve Position Indicator
10. Sodium Hydroxide Storage Tank Level

Table 4.3-7 shall be deleted.

The TS for the Reactor Building Area High Range Radiation Monitors, TS 3/4.3.3.1, Table 3.3-6 and 4.3-3, Item 1.c, and the Hydrogen Monitors, TS 3/4.6.5.1, have been relocated into this TS in order to have a consistent set of technical specifications.

The allowable outage time for the accident monitoring instruments has been changed from 7 to 30 days (for conditions in which the number of OPERABLE accident monitoring channels is less than the Required Number of Channels) and from 48 hours to 7 days (for conditions in which the number of OPERABLE accident monitoring channels is less than the Minimum Channels Operable), requiring a reduction to mode 3 (HOT STANDBY) within 6 hours and then to mode 4 (HOT SHUTDOWN) within the next 12 hours.

The mode APPLICABILITY for the Reactor Building Hydrogen Monitors was changed to include mode 3; the Reactor Building Area High Range Radiation Monitor was changed to delete mode 4.

Administrative changes were made to TS 3/4.3.3.6, Table 3.3-10.

BASIS FOR DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

SCE&G has evaluated the proposed TS change and has determined that it does not represent a significant hazards consideration, based on the criteria established in 10 CFR 50.92(c). Operation of VCSNS in accordance with the proposed action will not:

(1) Involve a significant increase in the probability or the consequences of an accident previously evaluated.

Regulatory Guide 1.97 furnishes standards acceptable to the NRC for providing instrumentation to monitor plant variables and systems during and following an accident. The purpose of the accident monitoring instrumentation is to display plant variables that provide information required by the control room operators for manual actions and long term recovery. Determination of variable types and category designations for VCSNS was accomplished from a review of the Emergency Response Guidelines (ERGs), the Final Safety Analysis Report, and the Westinghouse Owners Group (WOG) ERGs. The WOG ERGs were used at VCSNS as a basis for the Emergency Response Procedures. Operability of the instruments used for accident monitoring ensures there is sufficient information available on selected plant parameters to monitor plant status during and following an accident. The changes proposed do not affect components that can cause an accident. The increase in allowable outage times from 7 to 30 days or from 48 hours to 7 days does not significantly affect the consequences of an event previously evaluated. The channel redundancy and the relatively short outage times, coupled with the low probability of an event requiring accident monitoring instrumentation during this interval, ensure that sufficient information is available for operator manual actions. The condition of the plant in either HOT STANDBY or HOT SHUTDOWN, the first stage of the plant shutdown process, has no impact on the assumptions made in the accident analysis.

The change in mode applicability for the Reactor Building Area High Range Radiation Monitors to include modes 1, 2, and 3, but exclude mode 4, is based on the usage of these monitors which is to indicate a significant degradation of the reactor coolant pressure boundary. These monitors do not initiate any automatic mitigation system and are solely required to be operable to provide indication which in conjunction with other operator actions will aid in mitigating the consequences of design basis accidents. Design basis accident sequences which may create a significant degradation of the reactor coolant pressure boundary are not postulated to occur during mode 4. Therefore, the proposed change does not increase the probability or consequences of any accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change is consistent with the requirements of RG 1.97. The accident monitoring instrumentation will make available reliable information to plant control room operators to mitigate

the consequences of a design basis accident. The first stages of plant shutdown, HOT STANDBY and HOT SHUTDOWN, are plant modes for which VCSNS has been analyzed. Since no plant configuration changes or changes to the mode of operation of equipment, systems, and components are introduced by the proposed technical specification, no new failure modes or accident sequences are instituted. Therefore, the changes proposed do not create the possibility of a new or different kind of accident from any previously analyzed.

(3) Involve a significant reduction in a margin of safety.

The inclusion of category 1, type A or B, instrumentation in the TS provides assurance that adequate information is available to the operators to maintain VCSNS in a safe condition during and following a design basis accident. Accomplishment of specific manual action by the control room operators is enhanced due to the availability and reliability of the indications. The proposed changes do not affect the design or operation of safety related components relied upon to automatically mitigate the consequences of a design basis event. The proposed change from HOT SHUTDOWN to HOT STANDBY as the first stage of plant shutdown will not affect the design or operation of any safety related system or component. Therefore, the changes proposed would not involve a reduction in any margin of safety.