ATTACHMENT (2)

IMPROVED TECHNICAL SPECIFICATIONS, REVISION 13

REVISION BY IMPROVED TECHNICAL SPECIFICATION SECTION

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
March 16, 1998

Page Replacement Instructions
VOLUME 5
Section 3.1

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	INSERT
CTS Markup & Discussion of Changes	
DOC 3.1.7-1	DOC 3.1.7-1

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

ADMINISTRATIVE CHANGES

A.1 The proposed change will reformat, renumber, and reword the existing Technical Specifications, with no change of intent, to be consistent with NUREG-1432. As a result, the Technical Specifications should be more easily readable and, therefore, understandable by plant operators, as well as other users.

During the Calvert Cliffs ITS development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe each LCO and to be consistent with NUREG-1432. However, the additional information does not change the intent of the current Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to existing Specifications.

A.2 Current Technical Specification 3.10.1 SR 4.10.1.1 requires the position of each full length CEA required either partially or fully withdrawn be verified once per two hours. Improved Technical Specification SR 3.1.7.1 will add clarifying words that the positions be verified within the acceptance criteria for available negative reactivity addition. The CTS SR is performed in order to determine whether existing actions need to be taken. Therefore, adding clarifying words is an administrative change. This change is consistent with NUREG-1432.

A.3 Current Technical Specification 3.10.1 LCO states that the STE is for measurement of the CEA worth and SDM. Improved Technical Specification 3.1.7 will delete the requirement that the measurement is for SDM. The SDM is verified by being in compliance with the insertion limits of LCOs 3.1.5 and 3.1.6. Measurement of CEA worth is a measurement of the SDM which exists in a different LCO. Therefore, the elimination of redundancy constitutes an administrative change. This change is consistent with NUREG-1432, TSTF-67.

A.4 Current Technical Specification 3.10.1 requires the suspension of the SDM Specification during STE which measures CEA worth. Improved Technical Specification 3.1.7 will also suspend SDM, but in addition, will suspend the regulating CEA insertion limit and Shutdown CEA insertion limit Technical Specifications. This change is considered administrative because Calvert Cliffs currently suspends these additional Specifications by invoking another STE (CTS 3.10.2). Movement of a requirement within the Technical Specifications constitutes an administrative change. This change is consistent with NUREG-1432, CEOG-108.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - RELOCATIONS

None

Page Replacement Instructions
VOLUME 7
Section 3.3

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REMOVE	INSERT
ITS	
3.3.10-3	3.3.10-3
ITS Bases	
B3.3.10-19	B3.3.10-19
CTS Markup & Discussion of Changes	
DOC 3.3.1-1	DOC 3.3.1-1
DOC 3.3.7-1	DOC 3.3.7-1
DOC 3.3.7-3	DOC 3.3.7-3
Specification 3.3.10, Unit 1 Page 4 of 8	Specification 3.3.10, Unit 1 Page 4 of 8
Specification 3.3.10, Unit 2 Page 4 of 8	Specification 3.3.10, Unit 2 Page 4 of 8
DOC 3.3.10-1 through 3.3.10-7	DOC 3.3.10-1 through 3.3.10-7
ISTS Markup & Justification	
3.3-42	3.3-42
ISTS Bases Markup & Justification	
B 3.3-151	B 3.3-151
DOD 3.3-4 through 3.3-6	DOD 3.3-4 through 3.3-6

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SURVEILLANCE REQUIREMENTS

---NOTE---
These Surveillance Requirements apply to each PAM instrumentation Function in Table 3.3.10-1.

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK for each required indication channel that is normally energized.	31 days
SR 3.3.10.2	Perform a CHANNEL CALIBRATION on Containment Hydrogen Analyzers.	46 days on a STAGGERED TEST BASIS
SR 3.3.10.3	---NOTE--- Neutron detectors, Core Exit Thermocouples, and Reactor Vessel Level Monitoring System are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION on each indication channel except Containment Hydrogen Analyzers.	24 months

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BASES

channels are verified to be reading at the bottom of the range and not failed downscale.

For the Hydrogen Monitors, a CHANNEL CHECK is performed by
drawing a sample from the Waste Gas System through the 8
monitor.

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one indication channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.10.2

A CHANNEL CALIBRATION is performed every 46 days on a staggered test basis for the Containment Hydrogen Analyzers.
The CHANNEL CALIBRATION is performed using sample gases in
accordance with manufacturer's recommendations. 8

SR 3.3.10.3

A CHANNEL CALIBRATION is performed every 24 months or approximately every refueling. CHANNEL CALIBRATION is a check of the indication channel including the sensor. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note
allows exclusion of neutron detectors, Core Exit
Thermocouples, and Reactor Vessel Level Monitor System from 13
the CHANNEL CALIBRATION.

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is

ADMINISTRATIVE CHANGES

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During the Calvert Cliffs Improved Technical Specification (ITS) development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe Limiting Condition for Operation (LCO) and to be consistent with NUREG-1432. However, the additional information does not change the intent of the Current Technical Specification (CTS). The reformatting, renumbering, and rewording process involves no technical changes to existing Specifications.

A.2 Current Technical Specification 3.3.1.1 LCO requires a total number of reactor protective instrumentation channels and bypasses to be Operable, as required by Table 3.3-1. Improved Technical Specification LCO 3.3.1 requires four Reactor Protective System (RPS) bistable trip units and associated measurement channels and automatic bypass removal features of Table 3.3.1-1 to be Operable. This change essentially moves the total number of channels column from CTS Table 3.3-1 to the LCO. Moving requirements within a Technical Specification is an administrative change. This change is consistent with NUREG-1432.

A.3 A Note was added to CTS 3.3.1.1 which allows separate Condition entry for each RPS function. The Note in ITS 3.3.1 provides explicit instructions for proper application of the actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for the Reactor Protective System (RPS) Instrumentation. This change is consistent with NUREG-1432.

A.4 Current Technical Specification 3.3.1.1, Table 3.3.1, Note # to Action 2 exempts Specification 3.0.4 by stating it is not applicable. Exempting Specification 3.0.4 allows Modes to be changed while the LCO is not met. Improved Technical Specification 3.3.1 deletes this requirement when one channel is inoperable (Action A), but retains it when two are inoperable (Action B). This Note is only required when the required Actions do not allow continued Operation, therefore, it has been deleted for Action A and retained for Action B. Deleting a Note when it is not required constitutes an administrative change. This change is consistent with NUREG-1432.

A.5 A Note was added to CTS 3.3.1.1 Surveillances which requires that ITS Table 3.3.1-1 be referred to in order to determine which Surveillance Requirements (SRs) shall be performed for each RPS function. This Note is essentially an informational Note which clarifies that the specific Surveillances for each function are listed in ITS 3.3.1 Table 3.3.1-1. The addition of this informational Note constitutes an administrative change since no Technical Requirements were added. This change is consistent with NUREG-1432.

A.6 Current Technical Specification 3.3.1.1 Action 2.c requires an additional inoperable channel to be placed in the bypassed condition, provided the other channel is placed in the tripped condition. Improved Technical Specification 3.3.1 Action B will allow either channel to be

ADMINISTRATIVE CHANGES

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During the Calvert Cliffs ITS development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe each LCO and to be consistent with NUREG-1432. However, the additional information does not change the intent of the CTS. The reformatting, renumbering, and rewording process involves no technical changes to existing Specifications.

A.2 Current Technical Specification SR 4.3.2.1.1 requires testing of the logic for the bypasses. Improved Technical Specification 3.3.7 will delete this requirement for the Containment Purge Valve Isolation function because this function does not have any Technical Specification required bypasses. Deleting of a requirement that is not applicable constitutes an administrative change. This change is consistent with NUREG-1432.

A.3 Current Technical Specification 3.3.2.1 Table 3.3-3 contains a "Minimum Channels Operable" column. Improved Technical Specification 3.3.7 deletes this column because the Actions in the ITS are based on the number of channels inoperable, from the total required number of channels, which is specified in the LCO. This change is administrative in nature because the minimum channels Operable requirement is no longer used. This change is consistent with NUREG-1432.

A.4 Current Technical Specification 3.3.2.1 does not contain any Actions if the Required Actions and associated Completion Times cannot be met. Since the Specification is applicable in Mode 6, entering Specification 3.0.3 is not appropriate. Improved Technical Specification 3.3.7 would require the closure of the affected purge valves and the requirement to enter the applicable Condition(s) of LCO 3.9.3. This change is considered administrative because CTS Actions would require the closure of the containment purge valves. The requirement to enter LCO 3.9.3 is discussed in a less restrictive discussion of change. This change is consistent with NUREG-1432.

A.5 Current Technical Specification 3.3.2.1 Action a requires the associated Actions to be entered if the trip setpoint is less conservative than the allowable value. This is consistent with the Improved Technical Specification requirements which require Actions to be entered if the allowable value is exceeded because the channel is inoperable. Specifically stating this is not required. Therefore, the CTS requirement which states this is being deleted. This change is an administrative change. This change is consistent with NUREG-1432.

A.6 Current Technical Specification 3.3.2.1 Action b requires the Actions of Table 3.3-3 to be entered if an ESFAS instrumentation channel is inoperable. Improved Technical Specification 3.3.7 will delete this requirement because the Actions are no longer specified in the Instrumentation Table. The ITS Actions are specified in the Actions section of the Technical Specifications. The deletion of a requirement that is no longer applicable is considered an administrative change. This change is consistent with NUREG-1432.

change. This change will not have an adverse affect on plant safety. This change is consistent with NUREG-1432.

M.2 Not used.

M.3 Current Technical Specification 3.3.2.1 requires a CFT to be performed on the Containment Radiation Signal. Improved Technical Specifications SR 3.3.7.2 also requires a CFT to be performed. However, the ITS SR is modified by a Note which requires testing of actuation logic to include verification of the proper driver relay output signal. This is an additional requirement which is added to the Technical Specification SR, and is therefore a more restrictive change. This change will not adversely impact plant safety because verifying the proper relay driver output signal is currently included in the CFT. This change is consistent with NUREG-1432.

M.4 Improved Technical Specifications will add a Surveillance (SR 3.3.7.2) to perform a CFT on each Containment Radiation Signal Actuation Logic channel once per 92 days. Current Technical Specification 3.3.2.1 does not contain this SR. This SR will ensure that the containment purge valve isolation signal is properly tested. This Surveillance was added to CTS 3.3.2.1. The addition of new requirements to CTS constitute a more restrictive change. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - MOVEMENT OF INFORMATION TO LICENSEE-CONTROLLED DOCUMENTS

LA.1 Not used.

LA.2 Current Technical Specification 3.3.2.1 Table 3.3-3 contains a "Channels to Trip" column. Improved Technical Specification 3.3.7 will not contain this information. This is an informational column which is more appropriate for the Bases. The number of channels to trip will not be changed. The information is being moved to the Bases intact. Any changes to these requirements in the Bases will require change in compliance with the Bases Change Control Program in ITS Section 5.0. However, any hardware change to the number of channels to trip will require a design change. The Bases Change Control Program will ensure that changes receive appropriate review. This change is a less restrictive movement of details change with no impact on safety. This change is consistent with NUREG-1432.

Specification 3.3.10

~~TABLE 4.3-10~~ (A.1)

~~POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

INSTRUMENT	SR 3.3.10.1 CHANNEL CHECK	SR 3.3.10.3 CHANNEL CALIBRATION
7 ~~1~~. Containment Pressure	M	REFUELING INTERVAL
1 ~~2~~. Wide Range Logarithmic Neutron Flux Monitor	M	NA (A.4)
2 ~~3~~. Reactor Coolant Outlet Temperature	M	REFUELING INTERVAL
11 ~~4~~. Pressurizer Pressure	M	REFUELING INTERVAL
13 ~~5~~. Pressurizer Level	M	REFUELING INTERVAL
12 ~~6~~. Steam Generator Pressure	M	REFUELING INTERVAL
14 ~~7~~. Steam Generator Level (Wide Range)	M	REFUELING INTERVAL
~~8. Auxiliary Feedwater Flow Rate~~	~~M~~	~~REFUELING INTERVAL~~
4 ~~9~~. RCS Subcooled Margin Monitor	M	REFUELING INTERVAL
~~10. PORV/Safety Valve Acoustic Monitor~~	~~NA~~	~~REFUELING INTERVAL~~
~~11. PORV Solenoid Power Indication~~	~~NA~~	~~NA~~
~~12. Feedwater Flow~~	~~M~~	~~REFUELING INTERVAL~~
6 ~~13~~. Containment Water Level (Wide Range)	M	REFUELING INTERVAL
5 ~~14~~. Reactor Vessel Water level	M	NA
16-19 ~~15~~. Core Exit Thermocouple System	M	R
10. Containment Hydrogen Analyzers		46 day, Staggered Test Basis (A.1) SR 3.3.10.3
15. Condensate Storage Tank Level		
3. Reactor Coolant Inlet Temperature	M	24 months
8. Containment Isolation Valve Position	M	24 months
9. Containment Area Radiation (high range)	M	24 months (A.1) (LA.12)
20. Pressurizer Pressure (low range)	M	24 months (M.5)

(L.8) (A.6) (M.5)

SR 3.3.10.3 NOTE ~~*~~ The performance of a CHANNEL CALIBRATION operation exempts the Core Exit Thermocouple but includes all electronic components. The Core Exit Thermocouple shall be calibrated prior to installation in the reactor core.

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Specification 3.3.10

A.1

~~TABLE 4.3-10~~

~~POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

INSTRUMENT	CHANNEL CHECK (SR 3.3.10.1)	CHANNEL CALIBRATION (SR 3.3.10.2)
7 ~~1~~. Containment Pressure	M	REFUELING INTERVAL
1 ~~2~~. Wide Range Logarithmic Neutron Flux Monitor	M	NA (A.4)
2 ~~3~~. Reactor Coolant Outlet Temperature	M	REFUELING INTERVAL
11 ~~4~~. Pressurizer Pressure	M	REFUELING INTERVAL
13 ~~5~~. Pressurizer Level	M	REFUELING INTERVAL
12 ~~6~~. Steam Generator Pressure	M	REFUELING INTERVAL
1? ~~7~~. Steam Generator Level (Wide Range)	M	REFUELING INTERVAL
~~8. Auxiliary Feedwater Flow Rate~~	~~M~~	~~REFUELING INTERVAL~~
4 ~~9~~. RCS Subcooled Margin Monitor	M	REFUELING INTERVAL
~~10. PORV/Safety Valve Acoustic Monitor~~	~~NA~~	~~REFUELING INTERVAL~~
~~11. PORV Solenoid Power Indication~~	~~NA~~	~~NA~~
~~12. Feedwater Flow~~	~~M~~	~~REFUELING INTERVAL~~
6 ~~13~~. Containment Water Level (Wide Range)	M	REFUELING INTERVAL
5 ~~14~~. Reactor Vessel Water Level	M	NA
16–19 ~~15~~. Core Exit Thermocouple System	M	R(*)

L.8

A.6

SR 3.3.10.3 NOTE

(*) The performance of a CHANNEL CALIBRATION operation exempts the Core Exit Thermocouple but includes all electronic components. The Core Exit Thermocouple shall be calibrated prior to installation in the reactor core.

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10. Containment Hydrogen Analyzers		46 days Staggered Test Basis
15. Condensate Storage Tank Level	M	24 months
3. Reactor Coolant Inlet Temperature	M	24 months
8. Containment Isolation Valve Position	M	24 months
9. Containment Area Radiation (high range)	M	24 months
20. Pressurizer Pressure (low range)	M	24 months

A.1 SR 3.3.10.3

M.5

A.1

LA.12

M.5

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ADMINISTRATIVE CHANGES

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A.2 Improved Technical Specification 3.3.10 will reformat the applicable SRs of CTS 3.3.3.6 into the table format of, and consistent with, NUREG-1432. The reformatting of the applicable SRs alone does not constitute a change to the requirements for performing the applicable tests. Any other changes to the applicable requirements are identified in other discussion of changes.

A.3 A Note was added to CTS 3.3.3.6 which allows separate Condition entry for each function. The Note in ITS 3.3.10 provides explicit instructions for proper application of the actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for the Post-accident Monitoring (PAM) Instrumentation. This change is consistent with NUREG-1432.

A.4 Current Technical Specification 3.3.3.6 Table 4.3-10 does not require a channel calibration for the Wide Range Logarithmic Neutron Flux Monitor. Improved Technical Specification 3.3.10 will require a channel calibration for the wide range logarithmic neutron flux monitor. Although this requirement is being added to the PAM instrumentation ITS, it is currently required in CTS under the RPS instrumentation requirements, which utilize the same instrumentation. Therefore, this change is administrative since it does not add a new Surveillance; it only duplicates a requirement for a Surveillance that is currently performed as part of another Specification. This change is consistent with NUREG-1432.

A.5 Current Technical Specification 3.3.3.6 Actions 34 and 35-2 requires a special report to be submitted to the NRC within 30 days following the event, outlining the action taken, cause of the inoperability, and the plans and schedule for restoring the system to Operable status. Improved Technical Specification 3.3.10 Action F will require the immediate initiation of Action in accordance with Specification 5.6.7. The change essentially moves the reporting requirements for PAM instrument channel(s) inoperabilities to Chapter 5.0 of the proposed Specifications. The movement of this administrative requirement to Chapter 5.0 is an editorial preference, consistent with NUREG-1432.

A.6 Current Technical Specification 3.3.3.6 Table 4.3.10 contains an asterisk Note to the channel calibration requirement for the Core Exit Thermocouple System instrumentation. This Note exempts the core exit thermocouple from the channel calibration, but includes all electronic components and requires a calibration prior to installation. This Note will be incorporated into the Note for SR 3.3.10.3, Channel Calibration. The statement to include all electronic components will be deleted because the only exemption is for the LEG, and it is understood other components are not exempt, so the note is not needed. This change is classified administrative because the content of the first part of the Note has been incorporated into the Note for SR 3.3.10.3. In regards to the second part of the Note, SR 3.0.4 requires the performance of required Surveillances prior to the Mode of applicability of the LCO. Therefore, the requirement to ensure the CETs are calibrated prior to use is captured by SR 3.0.4. Since no substantive change is proposed, this change is administrative. This change is consistent with NUREG-1432.

A.7 Current Technical Specification 3.6.5.1 for the Hydrogen Analyzers is being incorporated into ITS 3.3.10, "PAM Instrumentation." Hydrogen is only a concern following an accident, and is a Type I PAM instrument. The movement of a Technical Specification requirement within the Technical Specifications constitutes an administrative change. This change is consistent with NUREG-1432.

A.8 Current Technical Specification SR 4.6.5.1.2 requires the hydrogen analyzer channel calibration to be performed once per 92 days on a Staggered Test Basis. Improved Technical Specifications will require the channel calibration to be performed once per 46 days on a Staggered Test Basis. This change is essentially equivalent because the definition of Staggered Test Basis requires both systems to be performed such that both systems are performed at two (the number of hydrogen analyzers) times the Surveillance Frequency (2 X 46 = 92 days). Therefore, this change is administrative. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Current Technical Specification 3.3.3.6 Actions 31 and 33 require the plant to be in Mode 4 in 12 hours, when two channels of the respective function are inoperable and cannot be restored to Operable status within the required time. Improved Technical Specification 3.3.10 will require an intermediate step for the plant to be in Mode 3 within 6 hours, and Mode 4 within 12 hours. Current Technical Specifications require the plant to be in Mode 4 within 12 hours, however no requirement for entry into Mode 3 is specified. The addition of this restriction, consistent with NUREG-1432, is an additional limitation on plant operation and, therefore, a more restrictive change. This change will not adversely affect plant safety. This change is consistent with NUREG-1432.

M.2 Current Technical Specification 3.3.3.6 Action 32, when one Containment Water Level (Wide Range) channel is inoperable, requires the channel be restored to Operable status at the next refueling outage of sufficient duration. Improved Technical Specifications 3.3.10 Actions A and B will also allow continued operation; however, if the channel cannot be restored within 30 days, a report is submitted to the NRC in accordance with 5.6.7. The addition of these new requirements constitutes a more restrictive change. The change will not adversely affect plant safety. This change is consistent with NUREG-1432.

M.3 Current Technical Specification 3.6.5.1 (Hydrogen Analyzers) contains a Mode of Applicability of Modes 1 and 2. Improved Technical Specification 3.3.10 will add Mode 3 to the Applicability consistent with the other PAM instrumentation. This was done because the possibility continues to exist that hydrogen could be present in containment following an accident in containment in Mode 3. The addition of another Mode to the Applicability constitutes a more restrictive change. The change will have no adverse affect on plant safety. This change is consistent with NUREG-1432. In conjunction with this change, the shutdown tract has been changed to add the requirement to be in Mode 4 in 12 hours. This accounts for the addition of Mode 3 to the Modes of Applicability.

M.4 Current Technical Specification 3.3.3.6, Actions 31 and 33, when two channels, of the respective functions are inoperable, requires one channel to be restored to Operable status in 30 days. Improved Technical Specification 3.3.10 Action C will require the channel to be restored to Operable status in 7 days. This change essentially reduces the allowed outage time from 30 days to 7 days. This change is acceptable because 7 days is enough time to restore one channel to Operable status. The change will not adversely affect plant safety. This change is consistent with NUREG-1432.

M.5 Current Technical Specification 3.3.3.6 Table 4.3-10 does not require the Reactor Coolant Inlet Temperature, Containment Isolation Valve Position, Condensate Storage Tank Level, or Pressurizer Pressure (low range) to be provided as PAM instruments in the CTS. These indications are Category I variables, and are listed as such in the Updated Final Safety Analysis Report, and all but Pressurizer Pressure (low range) in a letter from Baltimore Gas and Electric Company to the NRC, dated August 9, 1988, Regulatory Guide 1.97 Review Update. Pressurizer Pressure (low range) was added to the Updated Final Safety Analysis Report as part of Revision 19. This equipment is controlled appropriately to ensure proper operation. Adding these indication channels to the ITS is a more restrictive change because it was not in the CTS, but it is appropriate for PAM instrumentation and was already being properly controlled. The change will not adversely affect plant safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - MOVEMENT OF INFORMATION TO LICENSEE-CONTROLLED DOCUMENTS

LA.1 The CTS 3.3.3.6 Action 35-1 details, which describe compensatory measures that may be implemented when the Reactor Water Vessel Level instrumentation is inoperable, are being moved to the Bases. Details regarding operation or construction of equipment to meet Specifications are more appropriately located in the Bases. Any changes to the Bases will require compliance with the Bases Change Control Program in ITS Section 5.0. The Bases Change Control Program will ensure that any changes to these requirements will receive appropriate review. This change does not alter this requirement and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

LA.2 The details of the construction of the Reactor Vessel Water Level probes and what constitutes an Operable channel are being relocated to the Bases for ITS 3.3.10. Design details regarding operation or construction of equipment to meet Specifications are more appropriately located in the Bases. The requirement to maintain the function Operable is retained in LCO 3.3.10 which provides sufficient controls to assure that the appropriate margin of safety is maintained. Any change to the Bases will require compliance with ITS Section 5.0, Bases Control Program which will ensure that any changes will be appropriately reviewed. This change is consistent with NUREG-1432.

LA.3 Current Technical Specification 3.6.5.1 Action a.1 requires the verification of the containment atmosphere grab sampling capability when one hydrogen analyzer is inoperable. This alternate method of hydrogen detection is more appropriately contained in the Bases. This requirement is being moved to the Bases for ITS 3.3.10. Any change to the Bases will require compliance with the ITS Section 5.0 Bases Control Program will ensure that any changes will be appropriately reviewed. This change does not alter any requirement and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

LA.4 Current Technical Specification SR 4.6.5.1.1 requires that each hydrogen analyzer be demonstrated Operable at least biweekly, on a Staggered Test Basis, by drawing a sample from the Waste Gas System through the hydrogen analyzer. The details of how a CHANNEL CHECK is performed on these monitors is being moved to the Bases. Any change to the Bases will require compliance with the ITS 5.0 Bases Control Program which will ensure that any changes will be appropriately reviewed. The requirement to perform a CHANNEL CHECK is retained in ITS 3.3.10 and, therefore, this removal of detail has no impact on plant safety. This change is consistent with NUREG-1432.

LA.5 Current Technical Specification SR 4.6.5.1.2 requires that a channel calibration be performed on the hydrogen analyzer using sample gases in accordance with manufacturers' recommendations. Details of how a Surveillance is performed is not intended to be contained in the Technical Specifications. This requirement is being moved to the Bases of ITS 3.3.10. Any change to the Bases will require compliance with the ITS Section 5.0 Bases Control Program which will ensure that any changes will be appropriately reviewed. This change does not alter this requirement and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

LA.6 Not used.

LA.7 Not used.

LA.8 Not used.

LA.9 Not used.

LA.10 Not used.

LA.11 Not used.

LA.12 Not used.

LA.13 Not used.

LA.14 Current Technical Specification 3.3.3.1, Table 3.3.6, contains the measurement range for the Containment Area High Range monitors. The Containment Area High Range monitors are moved to ITS 3.3.10, Post Accident Monitoring Instrumentation. The measurement range is moved to the LCO Bases for 3.3.10. Any change to these requirements in the Bases will be controlled under the Technical Specification Bases Control Program which will ensure that an appropriate review is performed. This change is a less restrictive movement of details with no affect on safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 Current Technical Specification 3.3.3.6 Action 31 (for all the PAM instruments except Reactor Vessel Water Level and Containment Water Level) requires inoperable PAM instruments to be restored to Operable status in 30 days, or the plant must be in Hot Shutdown within the next 12 hours. Improved Technical Specification 3.3.10 will relax requirements applicable when a single channel of PAM instrumentation is inoperable, for CTS functions 1 through 12, and 15. The proposed change permits continued operation with one channel inoperable; however, after 30 days a report is required, in accordance with Chapter 5.0, "Administrative Controls" of the proposed Specifications. Post-accident monitoring provides indication and does not provide an active safety function. With one indication channel inoperable, another is available or, in the case of RCS Subcooled Margin Monitor (SMM), alternate means of indication are available. With the RCS SMM inoperable, many of the critical parameters monitored by the instrumentation can be verified by the core exit thermocouples and reactor vessel water level monitoring system. This change is consistent with NUREG-1432.

L.2 Current Technical Specification 3.3.3.6 Action 35, when two channels of Reactor Vessel Water Level instrumentation are inoperable, requires one channel to be restored to Operable status within 48 hours, or to submit a report to the NRC. Improved Technical Specification 3.3.10 will allow seven days to restore one inoperable channel prior to submitting a report to the NRC. This change essentially increases the time allowed to repair one inoperable channel from 48 hours to 7 days. This change is based on the other indication channel available, and the availability of other means to monitor level (e.g., core exit thermocouples). Extending the allotted time from 48 hours to 7 days before additional actions are required is a less restrictive change. This change is consistent with NUREG-1432.

L.3 Current Technical Specification 3.3.3.6 Action 35-3 requires the Reactor Vessel Water Level Monitoring System to be restored to Operable status at the next scheduled refueling outage if any channels are inoperable. Improved Technical Specification 3.3.10 will delete this requirement. Therefore, this change essentially eliminates the Technical Specification requirement for restoring the system to Operable status. However, the NRC will be informed of the schedule for restoring the system, as required by Specification 5.6.7. This requirement will ensure that the Reactor Vessel Water Level Monitoring System is restored to Operable status in a timely manner. Also, good operating practice and management oversight dictate that plant systems be restored as soon as possible. The deletion of a Completion Time to restore a piece of equipment constitutes a less restrictive change. This change is consistent with NUREG-1432.

L.4 Current Technical Specification 3.3.3.6 Action 34 requires the one inoperable reactor vessel water level monitoring channel to be restored to Operable status within 7 days, or submit a report to the NRC. Improved Technical Specification 3.3.10 will increase the Completion Time from 7 days to 30 days. The 30 days are reasonable based on availability of the other channel and the backup method of reactor vessel water level indication. This change is consistent with NUREG-1432.

L.5 Current Technical Specification 3.6.5.1 Action a.2, for one inoperable hydrogen analyzer, requires the unit to be in Mode 3 within 6 hours if the hydrogen analyzer cannot be restored to Operable status in 30 days, and if the required report is not submitted to the NRC within 60 days . Improved Technical Specification 3.3.10 would require an LCO 3.0.3 entry for this same situation. Limiting Condition for Operation 3.0.3 would require the unit to be in Mode 3 in 7 hours, and Mode 4 in 13 hours (shutdown to Mode 4 is required because the Applicability was extended to Mode 3). This change essentially increases the time to be in Mode 3 from 6 hours to 7 hours. This change is acceptable because of the other available hydrogen analyzer, the fact that the hydrogen analyzers do not provide an active function, and the remote possibility that the hydrogen analyzers would be needed in the one extra hour allowed to reach Mode 3. Allowing one extra hour to reach Mode 3 is considered a less restrictive change. This change is consistent with NUREG-1432.

L.6 Current Technical Specification 3.3.3.1, Table 3.3-6, requires that if the number of channels of Containment Area High Range radiation monitors is less than the minimum required, that the alternative monitoring method be initiated in 72 hours and they either be restored to Operable status within 7 days, or a special report be submitted with 30 days. Improved Technical Specification 3.3.10 will allow one indication channel to be inoperable for 30 days, after which action is immediately initiated in accordance with Specification 5.6.7 to specify the alternative methods which have been implemented and corrective actions planned. With two channels inoperable, the actions are similar to CTS 3.3.3.1, with seven days allowed to restore one channel to Operable status, and then taking action in accordance with Specification 5.6.7. This change is acceptable with one channel inoperable, the second channel is still available to provide indication, which is the same basis for all the other PAM instrumentation. In case two channels are inoperable, alternate indications such as grab samples are available. This change is consistent with NUREG-1432.

L.7 Current Technical Specification Table 3.3-6 and 4.3-3 require the Containment High Range Area Monitor to be operable in Modes 1-4. Current Technical Specification Table 4.3-4 also specifies an alarm/trip setpoint and the requirement for a channel functional test. The containment high range monitor provides indication of high radiation levels in containment under post accident conditions and also provides a closure signal to the containment vent/hydrogen purge line on detection of high radiation. For loss of coolant accidents, the containment vent/hydrogen purge line is automatically isolated by a SIAS signal as assumed in the maximum hypothetical accident. In the event of a fuel handling accident, the lower range containment radiation monitors also close the vent line. The lower range monitors are required operable by ITS LCO 3.3.7. Therefore, the requirements for channel functional test and alarm/trip setpoints are deleted since these requirements are only a backup to the SIAS signal and containment low range radiation monitor functions which are already retained in the ITS. Improved Technical Specification 3.3 10 will only require the containment area high range monitors to be operable in Modes 1, 2, and 3 to perform the post accident monitoring function consistent with the requirements for other post accident monitoring instruments.

L.8 Current Technical Specification 3.3.3.6, Table 3.3-10 and 4.3-10, PAM instrumentation requirements include the non-Category I variables Auxiliary Feedwater (AFW) Flow Rate, Power-Operated Relief Valve (PORV)/Safety Valve Acoustic Flow Monitoring, PORV Solenoid Power Indication, and Feedwater Flow. Category I and Type A variables, listed as such in both the Updated Final Safety Analysis Report and a letter from Baltimore Gas and Electric Company to the NRC, dated August 9, 1988, Regulatory Guide 1.97 Review Update, are appropriately controlled in the Technical Specifications. The requirements for control of non-Category I, non-Type A, variables have been deleted from the Technical Specifications. These non-Category I, non-Type A, variables are not necessary to perform diagnosis specified in the emergency procedures or to take preplanned manual actions for which no automatic control is provided as described by Regulatory Guide 1.97. The variables which do perform these functions are retained in ITS 3.3.10. This change does not alter this requirement and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

L.9 Current Technical Specification 3.3.3.1, Table 4.3-3, requires a shiftly Channel Check of the Containment Area High Range radiation monitors. Improved Technical Specification requires a monthly Channel Check for PAM instruments. These instruments perform no automatic mitigative function for PAM, and are only required to provide post accident indication of containment radiation levels. They do provide a closure signal to the containment vent/hydrogen purge line on detection of high radiation as a backup for SIAS, but this is not their PAM function, and SIAS is the function credited for shutting the valves. For purposes of PAM instrumentation, a monthly channel check is adequate and is consistent with the Current Technical Specification 3.3.3.6 requirements for all other PAM instruments. This change does not alter the requirement for the instruments to be operable to support post accident indication and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

L.10 Current Technical Specification 4.6.5.1.1 requires a bi-weekly channel check of the hydrogen monitors on a staggered test basis. This effectively translates to a channel check on one monitor each week. Improved Technical Specification requires a monthly channel check for PAM instruments. These instruments perform no automatic mitigative function and are only required to provide post accident indication of containment hydrogen levels. For purposes of PAM instrumentation, a monthly channel check is adequate and is consistent with the Current Technical Specification 3.3.3.6 requirements for all other PAM instruments. This change does not alter the requirement for the instruments to be operable to support post accident indication and, therefore, has no impact on plant safety. This change is consistent with NUREG-1432.

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------
These ~~SRs~~ Surveillance Requirements apply to each PAM instrumentation Function in Table 3.3.10-1. 11

SURVEILLANCE		FREQUENCY
<4.3.3.6> SR 3.3.10.1	Perform CHANNEL CHECK for each required instrumentation indication channel that is normally energized. 4	31 days
<4.3.3.6> SR 3.3.10.3	------------NOTE------------ Neutron detectors and Reactor Vessel Level Monitoring System Core Exit Thermocouples are excluded from CHANNEL CALIBRATION. ---------------------------- Perform CHANNEL CALIBRATION on each indication channel except Containment Hydrogen Analyzers. 20 21 13	~~18~~ 24 months 1
<4.6.5.1.2> SR 3.3.10.2	Perform a CHANNEL CALIBRATION on Containment Hydrogen Analyzers 21	46 days on a STAGGERED TEST BASIS 11

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (continued) 10

The Frequency of 31 days is based upon plant operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.11.2 10 3

A CHANNEL CALIBRATION is performed every 18 24 months or approximately every refueling. indication CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy. A Note allows exclusion of neutron detectors, Core Exit Thermocouples and Reactor Vessel Level Monitoring System from the CHANNEL CALIBRATION. 1 2 3 4 13

~~[At this unit, CHANNEL CALIBRATION shall find measurement errors are within the following acceptance criteria:]~~ 1

~~For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may 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 a gamma source.~~ 3

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an 18 24 month calibration interval for the determination of the magnitude of equipment drift. 1

SR 3.3.10.2

A CHANNEL CALIBRATION is performed every 46 days on a STAGGERED TEST BASIS for the Containment Hydrogen Analyzers. The CHANNEL CALIBRATION is performed using sample gases in accordance with manufacturer's recommendations 4 8

REFERENCES

1. Letter to USNRC from R.E. Denton, BGE, dated June 1, 1995
2 ~~1~~. ~~Plant specific document (e.g., FSAR, NRC Regulatory Guide 1.97, SER letter).~~ August 9, 1988, J. Tiernan memo to NRC, Regulatory Guide 1.97 Update. 6
3 ~~2~~. Regulatory Guide 1.97.
4 ~~3~~. NUREG-0737, Supplement 1.
5 ~~4~~. ~~NRC Safety Evaluation Report (SER).~~ UFSAR, Chapter 7 6

20. The SR 3.3.11.2 Note which excludes the Neutron Detectors from the channel calibration will be modified to also exclude the Core Exit Thermocouples and Reactor Vessel Level Monitoring System. Core Exit Thermocouples are calibrated prior to installation. There is no mechanism at Calvert Cliffs to perform a channel calibration on the Reactor Vessel Level Monitoring System. The instruments are usually replaced. This change is consistent with the Calvert Cliffs current licensing basis.

21. Surveillance Requirement 3.3.10.2 was added to require performance of a channel calibration on containment hydroger analyzers every 46 days on a Staggered Test Basis, instead of 24 months. This change is consistent with the Calvert Cliffs current licensing basis. In conjunction with the change a statement was added to SR 3.3.10.3 excluding the hydrogen analyzers.

22. This change adds Steam Generator Pressure and RCS Subcooled Margin Monitor (SMM) and Pressurizer Pressure (low range) to Table 3.3.11-1. This change is consistent with the Calvert Cliffs current licensing basis. Only one channel of RCS SMM is required, therefore, the referenced action on STS Table 3.3.11-1 is not applicable. Conditions A and B are only necessary for this function. Only one channel is required in the licensing basis due to diverse indication or core cooling from core exit thermocouples and reactor vessel water level.

23. This change removes Auxiliary Feedwater Flow from Table 3.3.11-1, removing PAM instrumentation. Auxiliary Feedwater Flow does not meet the threshold for inclusion in this PAM table. This change is consistent with the Calvert Cliffs current licensing basis.

24. This change excludes SR 3.3.1.8, Channel Calibration from the Loss of Load function. The Loss of Load Function cannot be calibrated because it senses a turbine trip and does not contain setpoints. Loss of Load function is verified operable by performance of a channel functional test. This change is consistent with the Calvert Cliffs current licensing basis.

25. Not used.

26. NUREG-1432 Specification 3.3.4 condition D requires an inoperable automatic bypass removal channel to be disabled or placed in trip or bypass in 1 hour. If the module is bypassed or tripped, the actions subsequently require the channel to be restored to operable status or placed in trip in 48 hours. ITS 3.3.4 condition C will require the inoperable block removal feature for a sensor block module to be disabled in 1 hour or placed in bypass. The assumed safety function of the sensor block module is to ensure that the ESFAS features which have bypass capability are automatically unblocked. With the unit disabled or bypassed, the ability to block an ESFAS function on that module is removed, i.e., disabling or bypassing the block module effectively removes it from the circuitry and fulfills the auto-unblock function. This action places the function in a one-out-of-three logic which ensures the required redundancy. This action is equivalent to continued operation with an RPS channel placed in trip permitted by LCO 3.3.1 Condition A.

27. This change adds a Condition (Condition E) and related Required Actions to address the different Mode requirements associated with ESFAS Actuation Logic channels of Specification 3.3.5. This change also adds a statement identifying the second Condition of Condition D as being for the Manual Actuation Channel. The two different Conditions are

required to address the different times at which the conditions leave the Modes of Applicability. This change is consistent with the Calvert Cliffs current licensing basis.

28. Response Time Testing will be added to Specification 3.3.9, "Chemical and Volume Control System Isolation Signal." The response time is currently tested for this signal at Calvert Cliffs and was therefore added to the Calvert Cliffs ITS. This change is consistent with the Calvert Cliffs current licensing basis.

29. This change makes the channel check for the Rate of Change of Power - High RPS trip only applicable to the Wide Range Logarithmic Neutron Flux Monitor. The channel check can only be performed on that equipment, and this change is consistent with the Calvert Cliffs current licensing basis.

30. Specification 3.3.4 will not include a Containment Radiation - High function for ESFAS because Calvert Cliffs does use Containment Radiation - High for ESFAS. This change is consistent with Calvert Cliffs design.

31. NUREG-1432 Specification 3.3.3 Conditions B and C contain Notes which state the Reactor Trip Circuit Breakers (RTCBs) associated with one inoperable channel may be closed for up to one hour for the performance of an RPS Channel Functional Test. Calvert Cliffs ITS 3.3.3 Conditions B and C will not contain this Note. The Note is not required because LCO 3.0.5 will allow the RTCBs associated with the inoperable channel to be closed to perform testing. Limiting Condition for Operation 3.0.5 allows equipment that has been removed from service or declared inoperable to comply with Actions to be returned to service under administrative control to perform testing required to demonstrate its Operability or Operability of other equipment.

32. Specification 3.3.3 will not include a Condition Note which allows three Matrix Logic channels to be inoperable due to a common power source de-energizing three matrix power supplies. The Note is not needed because if a power supply is lost, three matrix logic channels are de-energized and one trip leg is lost (four RTCBs are open in one trip leg). This places the plant in a safe condition. If a power supply fails and the RTCBs do not open, an operability determination must be made to determine the inoperability. It is not necessarily a Matrix Logic problem. This change is consistent with Calvert Cliffs' design.

33. NUREG-1432 Specification 3.3.1 provides an Action (Action F) which requires a shutdown to MODE 3 if the Required Actions and associated Completion Times are not met. Improved Technical Specification 3.3.1 separates this Action into two different Actions, Actions F and G, based on the Modes of Applicability of the associated functions. The Loss of Load and Axial Power Distribution-High functions may be bypassed below 15% RTP and are not required to be operable. Therefore, NUREG Action F is retained for all other functions and a separate Action (Action G) is added which only requires a power reduction below 15% RTP consistent with the requirements for operability of the Loss of Load and Axial Power Distribution-High functions.

34. NUREG-1432 Specification 3.3.2 requires the Rate of Change of Power-High RPS trip function to be Operable during shutdown. Improved Technical Specification 3.3.2 also requires the automatic bypass removal feature associated with this function to be Operable as part of the LCO statement. Current Technical Specification 4.3.1.1.2 requires the bypass functions to be

demonstrated operable and CTS Table 3.3-1 footnote d indicates that this function has an associated bypass feature. The NUREG provides actions for an inoperable bypass feature and also provides surveillance requirements for demonstrating the function Operable. This change maintains consistency within the ITS and is consistent with the current Technical Specification requirements.

35. NUREG-1432 Surveillance Requirement 3.3.3.1 requires quarterly testing of the RTCBs. CTS Table 4.3-1 requires the RTCBs be tested on a monthly basis. Calvert Cliffs has retained the existing more frequent testing of the RTCBs consistent with the licensing basis and CEN 327, "RPS/ESFAS Extended Test Internal Evaluation," June 2, 1986, including Supplement 1, dated March 3, 1989.

36. Not used.

37. NUREG-1432 Condition B of LCO 3.3.7 requires placing the purge valves in the closed position and entering the applicable conditions of LCO 3.6.3 for the inoperable purge valves which result from an inoperable manual actuation or automatic actuation channel. These actions are in error and have been corrected in ITS 3.3.7 to require either the valves be closed, or the actions of LCO 3.9.3 be entered for inoperable valves. The STS reference to LCO 3.6.3, Containment Isolation Valves, is inappropriate since that LCO applies in Modes 1, 2, 3, and 4 whereas, LCO 3.3.7 is applicable during core alterations and movement of fuel. LCO 3.9.3, however, has the same applicability of LCO 3.3.7 and is the appropriate reference. The STS is also in error to require both actions be accomplished since the actions are somewhat redundant. Closure of the purge valves performs the safety function of the automatic and manual functions covered by the LCO. The actions of LCO 3.9.3 would require that core alterations and fuel movement be suspended which removes the unit from the mode of applicability. Therefore, ITS 3.3.7 Condition B replaces the "and" requirement in the STS with an "or" to indicate that these actions accomplish the same desired result. This change is also consistent with the requirements of the STS for each of the other PWR owners groups.

38. The Applicability of NUREG LCO 3.3.1 is MODES 1 and 2. The ITS Applicability references Table 3.3.1-1, and Table 3.3.1-1 has a column specifying the Applicability for each Function in the Table. This method is necessary since each Function does not have the same Applicability; two of the Functions are only required in MODE 1 $\geq$ 15% RTP. This change is also consistent with the Calvert Cliffs CTS Table 3.3-1.

39. NUREG-1432 SR 3.3.9.2, Note 2, requires testing of relays after 24 hours of Mode 5 operation for relays that cannot be tested during operation. These relays would cause a closure of the letdown line which is undesirable during unit operation. The CVCS actuation logic is currently not required by the Technical Specifications. The 24 month test frequency proposed in ITS SR 3.3.9.2 is consistent with the CTS Table 4.3-2 footnotes 2-6 for other ESFAS functions actuation logic which cannot be tested online. The requirements of the STS would impose an unnecessary more restrictive change on plant operations and are not adopted.

40. The ESFAS Functions listed in Table 3.3.4-1 include trip and bypass removal features as appropriate. Referring to trip and bypass removal features as separate Functions is incorrect and confusing. Removing the words "trip or bypass removal" satisfies the intent of the Note and eliminates the error.

Page Replacement Instructions
VOLUME 8
Section 3.4

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	INSERT
CTS Markup & Discussion of Changes	
Specification 3.4.10, Unit 2	
Page 2 of 2	Page 2 of 2

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

INSERT APPLICABILITY NOTE

---NOTE---

The lift settings are not required to be within Limiting Condition for Operation limits during MODE 3 > 365 F for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 > 365 F provided a preliminary cold setting was made prior to heatup.

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301 301

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Page Replacement Instructions
VOLUME 10
Section 3.6

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	INSERT
CTS Markup & Discussion of Changes	
Specification 3.6.2, Unit 1	Specification 3.6.2, Unit 1
Page 1 of 7	Page 1 of 7
Specification 3.6.2, Unit 2	Specification 3.6.2, Unit 2
Page 1 of 7	Page 1 of 7
DOC 3.6.2-2 through 3.6.2-5	DOC 3.6.2-2 through 3.6.2-5

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

3/4.6 CONTAINMENT SYSTEMS

(A.1)

3/4.6.1 PRIMARY CONTAINMENT

Containment Air Locks

LIMITING CONDITION FOR OPERATION

LCO 3.6.2

3.6.1.3 ~~Each~~ TWO containment air locks shall be OPERABLE with: (A.2)

a. ~~Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and~~ (LA.1)

(A.6) 13

b. An overall air lock leakage rate of ≤ 0.05 L_a ~~(17,300 SCCM)~~ at P_a, ~~50 psig~~, as specified in Specification 6.5.6, "Containment Leakage Rate Testing Program." (A.1) Lic. Amend. 219

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

INSERT ACTION A, B, C (L.1) (M.1)

Actions A, B, & C

a. ~~With an air lock inoperable, except as a result of an inoperable door gasket, restore the air lock to OPERABLE status within 24 hours~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Action D

Action A. (inop door gasket)

~~b. With an air lock inoperable due to an inoperable door gasket:~~

~~1. Maintain the remaining door of the affected air lock closed and sealed, and~~ (L.2) (L.4)

~~2. Restore the air lock to OPERABLE status within 7 days~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Action D

INSERT ACTION NOTES: 1. (L.3) 2. (A.3) 3. (A.4)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. ~~After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours by verifying that the seal leakage is < 0.0002 L_a (69.2 SCCM) as determined by precision flow measurement when the volume between the door seals is pressurized to a constant pressure of 15 psig,~~ License Amend. 219

By performing containment air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program, and

~~* Exemption to 10 CFR Part 50, Appendix J.~~

DELETE

INSERT Notes 1 and 2 to SR 3.6.2.1 (A.5)

License Amendment 219

Specification 3.6.2

3.6 ~~3/4.6~~ CONTAINMENT SYSTEMS

3.6.2 ~~3/4.6.1~~ ~~PRIMARY CONTAINMENT~~ (A.1)

Containment Air Locks

LIMITING CONDITION FOR OPERATION

LCO 3.6.2 ~~3.6.1.3~~ ~~Each~~ Two containment air locks shall be OPERABLE with: (A.2)

~~a.~~ ~~Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and~~ (LA.1)

~~b.~~ An overall air lock leakage rate of ≤ 0.05 L_a ~~(17,300 SCCM) at P_a, (50 psig).~~ as specified in Specification ~~6.5.6~~ "Containment Leakage Rate Testing Program." (A.6) (13) License Amend. 196

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

INSERT ACTION A, B (L.1), C (M.1)

Actions A, B, C / Action D — a. ~~With an air lock inoperable, except as a result of an inoperable door gasket, restore the air lock to OPERABLE status within 24 hours~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Action A' (inop door gasket) — ~~b. With an air lock inoperable due to an inoperable door gasket:~~

~~1. Maintain the remaining door of the affected air lock closed and sealed, and~~ (L.2) (L.4)

Action D — ~~2. Restore the air lock to OPERABLE status within 7 days~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT ACTION NOTES 1 (L.3), 2 (A.3), 3 (A.4)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 — 4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. ~~After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours by verifying that the seal leakage is < 0.0002 L_a (69.2 SCCM) as determined by precision flow measurement when the volume between the door seals is pressurized to a constant pressure of 15 psig.~~

By performing containment air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program, and

License Amendment 196

INSERT Notes 1 and 2 to SR 3.6.2.1 — (A.5)

Delete — ~~* Exemption to 10 CFR Part 50, Appendix J.~~

page 1 of 7

License Amendment 196

event a single air lock door was inoperable. Additionally, the results of the air lock leakage test are accounted for in determining the combined Type B and C containment leakage rate. The addition of these Notes is administrative, because it does not change the intent of the CTS.

A.6 Current Technical Specification LCO 3.6.1.3.a requires an overall air lock leakage rate of ≤ 0.05 L_a. Improved Technical Specification 3.6.2 will only require the overall air lock leakage to be in accordance with the Containment Leakage Rate Testing Program. The specific value of ≤ 0.05 L_a will be moved to ITS 5.5.16, Containment Leakage Rate Testing Program. Movement oft his requirement from CTS LCO 3.6.1.3.a to ITS 5.5.16 constitutes an administrative change. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Current Technical Specification 3.6.1.3 Action a requires the air lock to be restored to Operable status within 24 hours when an air lock is inoperable, except when the air lock is inoperable as a result of an inoperable door gasket. Improved Technical Specification 3.6.2 Action C is for air lock inoperabilities other than an inoperable air lock door and the interlock mechanism. This Action requires action to be immediately initiated to evaluate overall containment leakage rate, verification a door is closed in the affected air lock within 1 hour, and restoration of the air lock to Operable status within 24 hours. This change adds two additional requirements when the air lock is inoperable for reasons other than for an inoperable door and interlock mechanism. The Action to evaluate the overall containment leakage will determine if the containment is Operable. If it is determined the overall containment leakage exceeds the required limits, then the Actions of LCO 3.6.1 are entered (one hour to restore). The requirement to maintain at least one door closed is consistent with the Actions of LCO 3.6.1, which requires the containment to be restored to Operable status within one hour. Adding additional requirements to the Technical Specifications constitutes a more restrictive change. This change will not impact plant safety because the additional Actions were added to ensure that the containment is Operable. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - MOVEMENT OF INFORMATION TO LICENSEE-CONTROLLED DOCUMENTS

LA.1 Current Technical Specification LCO 3.6.1.3.a requires both doors to be closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed. This requirement is being moved out of Technical Specifications and into ITS Bases 3.6.2. This requirement is consistent with the containment air lock interlock mechanism which prevents both doors from being open at one time. The containment air lock interlock is required to be Operable consistent with this requirement in the CTS. The requirement that only one door be opened at a time in the containment air lock will still be ensured, although this information is being moved to the Bases. Informational details such as this are consistently being moved to the Bases as part of the conversion to NUREG-1432. Any changes to this information in the Bases will be consistent with the requirements of the Bases Control Program of Section 5.0. The Bases Control Program will ensure that changes to these requirements will be appropriately reviewed. The level of safety of facility operation is unaffected by the change because there is no change in the requirement to maintain the containment air lock interlock Operable, which will prevent both air lock doors from being open. Furthermore, NRC and Calvert Cliffs resources associated with processing license amendments to these requirements will be reduced. This change is a less restrictive movement of information change with no impact on safety. This change is consistent with NUREG-1432.

LA.2 Not used.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 Current Technical Specification 3.6.1.3 Action a requires the air lock to be restored to Operable status within 24 hours when an air lock is inoperable, except when the air lock is inoperable as a result of an inoperable door gasket. Improved Technical Specification Actions A (one or more air locks with one containment air lock door inoperable) and B (one or more containment air locks with containment air lock interlock mechanism inoperable) will require the following Actions:

1. Within 1 hour verify an Operable door is closed in the affected air lock;
2. Within 24 hours lock the Operable door closed in the affected air lock; and
3. Once per 31 days verify the Operable door is locked closed in the affected air lock (this Action is modified by a Note which allows the air lock doors in high radiation areas to be verified locked closed by administrative means).

This change increases the time to restore an inoperable air lock door to Operable status (when not due to a gasket being inoperable), or return the interlock mechanism to Operable status from 24 hours to no time limit as long as the Operable door is locked closed and verified closed periodically. This change is acceptable because when an Operable air lock door is maintained closed, a leak tight barrier separates containment from the outside atmosphere. This ensures that no radioactive material will be released through the air locks. Relaxing the Completion Time to restore the air lock to Operable status constitutes a less restrictive change. This change is consistent with NUREG-1432.

L.2 Current Technical Specification 3.6.1.3 Action b, when an air lock is inoperable as a result of an inoperable door gasket, requires the Operable door to be maintained locked closed and sealed, and the air lock restored to Operable status within 7 days. Improved Technical Specification Action A will require the following Actions:

1. Within 1 hour verify an Operable door is closed in the affected air lock;
2. Within 24 hours lock the Operable door closed in the affected air lock; and
3. Once per 31 days verify the Operable door is locked closed in the affected air lock (this Action is modified by a Note which allows the air lock doors in high radiation areas to be verified locked closed by administrative means).

This change increases the time to restore an inoperable air lock door to Operable status (when due to a gasket being inoperable) from seven days to no time limit as long as the Operable door is locked closed and verified closed periodically. This change is acceptable because when an Operable air lock door is maintained closed, a leak tight barrier separates containment with the outside atmosphere. This ensures that no radioactive material will be released through the air locks. Relaxing the Completion Time to restore the air lock to Operable status constitutes a less restrictive change. This change is consistent with NUREG-1432.

L.3 Improved Technical Specification 3.6.2 contains an Actions Note (Note 1) which allows entry and exit to perform repairs on the affected air lock components. Current Technical Specification 3.6.1.3 Action b does not contain this allowance, although Action b (air lock inoperable due to an inoperable gasket) does contain a requirement to maintain the Operable air lock door in a closed position. This change will allow an exception to the Actions to maintain the Operable door in a closed position by allowing it to be opened to perform repairs on the inoperable gasket. If the outer door is inoperable, then it may be easily accessed for most repairs (the inner door is providing the leak tight barrier). If the inner door is inoperable, it is preferred (as stated in the Bases) that the air lock be accessed from inside containment by entering through the other air lock. If this is not practicable, or if repairs on either door must be performed from the barrel side of the door, then it is permissible to enter the air lock through the Operable door. After each entry and exit, the Operable door must be immediately closed. This is acceptable because of the short time period during which the containment boundary is not intact and the low probability of an event occuring that would pressurize containment during this period. The addition of an allowance which allows entry and exit through an inoperable air lock constitutes a less restrictive change. This change is consistent with NUREG-1432.

L.4 Improved Technical Specification 3.6.2 contains a Note (Note 2) in Action A which allows entry and exit into containment for seven days under administrative controls if both air locks are inoperable due to each having an inoperable door. Current Technical Specification 3.6.1.3 Action b does not contain this allowance. The seven-day allowance begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specification SRs and Required Actions, as well as other activities on equipment inside containment that support Technical Specification required equipment. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the Operable door is expected to be open. This change is consistent with NUREG-1432.

L.5 Current Technical Specification 4.6.1.3.c requires verification that only one door in each air lock can be opened at a time (interlock mechanism), once per six months. Improved Technical Specification SR 3.6.2.2 will require this Surveillance to be performed once per 24 months. This change decreases the Frequency for verification of the interlock mechanism from 6 months to 24 months. This test ensures that the interlock, which prevents both air lock doors from being opened at the same time, is Operable. Since closure of either door will support containment Operability, normal personnel entry and exit will maintain containment integrity. The 24-month Frequency is based on the potential for loss of containment Operability if the SR were to fail when being performed with the reactor at power. It also is based on engineering judgment and is considered adequate given that the interlock is not typically challenged during use of the air locks. Decreasing SR Frequencies constitutes a less restrictive change. This change is consistent with NUREG-1432.

Page Replacement Instructions
VOLUME 11
Section 3.7

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	INSERT
ITS Bases	
B 3.7.10-1	B 3.7.10-1
ISTS Bases Markup & Justification	
B 3.7-66	B 3.7-66

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

B 3.7 PLANT SYSTEMS

B 3.7.10 Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)

BASES

BACKGROUND

The ECCS PREFS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA).

The ECCS PREFS consists of two independent and redundant 6
fans, a prefilter, a high efficiency particulate air (HEPA) 1
filter, and an activated charcoal adsorber section for
removal of gaseous activity (principally iodines). 6
Ductwork, valves or dampers, and instrumentation also form 1
part of the system.

The ECCS PREFS operates during normal unit operations.
During normal operation flow goes through the pre-filter and
HEPA filters, but flow through the charcoal adsorbers is 6
bypassed. During emergency operations, the ECCS PREFS dampers are realigned to initiate filtration. The stream of ventilation air discharges through the system filter trains and out the plant stack. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREFS is discussed in the Updated Final Safety Analysis Report (UFSAR), Section 9.8.2.3 (Ref. 1), as it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

Emergency Core Cooling System PREFS ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment as a layer of defense.

13

2

BASES

9.8.2.3

BACKGROUND (continued)

The ECCS PREACS is discussed in the FSAR, Sections ~~[6.5.1]~~, ~~[9.4.5], and [15.6.5]~~ (Refs. 1, ~~2, and 3, respectively~~), as it may be used for normal, as well as post accident, atmospheric cleanup functions. ~~The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies, as discussed in the Regulatory Guide 1.52 (Ref. 4).~~ 6

APPLICABLE SAFETY ANALYSES

ECCS PREFS ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment as a layer of defense.

~~The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as safety injection pump seal failure, during the recirculation mode. In such a case, the system limits the radioactive release to within 10 CFR 100 limits (Ref. 5), or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large break LOCA is presented in Reference 3. The ECCS PREACS also actuates following a small break LOCA, requiring the unit to go into the recirculation mode of long term cooling and to clean up releases of smaller leaks, such as from valve stem packing.~~ 3 1

~~The two types of system failures that are considered in the accident analysis are complete loss of function and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.~~ 3 6 13

The ECCS PREACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

~~Two independent and redundant~~ ECCS PREACS ~~trains~~ ~~are~~ is required to be OPERABLE to ensure that ~~at least one is available, assuming a single failure disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding the required limits in the event of a Design Basis Accident (DBA).~~ 3 1

(continued)

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VOLUME 12
Section 3.8

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	INSERT
ITS	
3.8.1-11 and 3.8.1-12	3.8.1-11 and 3.8.1-12
ITS Bases	
B 3.8.1-3	B 3.8.1-3
CTS Markup & Discussion of Changes	
DOC 3.8.1-1	DOC 3.8.1-1
ISTS Bases Markup & Justification	
B 3.8-61	B 3.8-61

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.8.1.4	------------------- NOTES ------------------- 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients below the load limit do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This Surveillance Requirement shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.3 or SR 3.8.1.9. --		9
	Verify each DG is synchronized and loaded, and operates for ≥ 60 minutes at a load ≥ 4000 kW for DG 1A and ≥ 2700 kW for DGs 1B, 2A, and 2B.	31 days	13
SR 3.8.1.5	Verify each day tank contains ≥ 325 gallons of fuel oil for DG 1A and ≥ 275 gallons of fuel oil for DGs 1B, 2A, and 2B.	31 days	
SR 3.8.1.6	Check for and remove accumulated water from each day tank.	31 days	
SR 3.8.1.7	Verify the fuel oil transfer system operates to automatically transfer fuel oil from storage tank[s] to the day tank.	31 days	9 11

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.8.1.8	Verify interval between each sequenced load block is within ± 10% of design interval for the load sequencer.	31 days	
SR 3.8.1.9	--------------------NOTE-------------------- All DG starts may be preceded by an engine prelube period. --		
	Verify each DG starts from standby condition and achieves, in ≤ 10 seconds, voltage > 3740 V and frequency > 58.8 Hz, and after steady state conditions are reached, maintains voltage > 3740 V and ≤ 4580 V and frequency of > 58.8 Hz and ≤ 61.2 Hz.	184 days	9
SR 3.8.1.10	Verify manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit.	24 months	9
SR 3.8.1.11	--------------------NOTE-------------------- Momentary transients outside the load and power factor limits do not invalidate this test. --		9
	Verify each DG, operating at a power factor of ≤ 0.85, operates for ≥ 60 minutes while loaded to ≥ 4000 kW for DG 1A and ≥ 3000 kW for DGs 1B, 2A, and 2B.	24 months	9 13
SR 3.8.1.12	Verify each DG rejects a load ≥ 500 hp without tripping.	24 months	9

BASES

injection actuation signal or on a 4.16 kV degraded or undervoltage signal. If both 4.16 kV offsite source breakers are open, the DG, after reaching rated voltage and frequency, will automatically close onto the 4.16 kV bus.

In the event of a loss of offsite power to a 4.16 kV 1E bus, if required, the ESF electrical loads will be automatically sequenced onto the DG in sufficient time to provide for safe shutdown for an anticipated operational occurrence (AOO) and ensure that the containment integrity and other vital functions are maintained in the event of a design bases accident.

Ratings for the 1A DG satisfies the requirements of Regulatory Guide 1.9 (Ref. 3) and ratings for the 1B, 2A, and 2B DG satisfy the requirements of Safety Guide 9 (Ref. 4). The continuous service rating for the 1A DG is 5400 kW and for the 1B, 2A, and 2B DGs are 3000 kW. 13

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 5) and Chapter 14 (Ref. 6), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the

ADMINISTRATIVE CHANGES

A.1 The proposed change will reformat, renumber, and reword the existing Technical Specifications, with no change of intent, to be consistent with NUREG-1432. As a result, the Technical Specifications should be more easily readable and, therefore, understandable by plant operators, as well as other users.

During the Calvert Cliffs Improved Technical Specification (ITS) development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe each Limiting Condition for Operation (LCO) and to be consistent with NUREG-1432. However, the additional information does not change the intent of the CTS. The reformatting, renumbering, and rewording process involves no technical changes to existing Specifications.

A.2 Current Technical Specifications (CTS) 3.8.1.1.a requires two "physically independent" offsite circuits to be Operable. Improved Technical Specification 3.8.1 will require two "qualified" offsite circuits to be Operable. The ITS use of qualified (as defined in the Bases) refers to the those offsite circuits that are described in the Updated Final Safety Analysis Report and are part of the licensing basis for the plant. Although qualified expands the description of offsite circuits, it does not technically change the requirement for the offsite circuits. The Updated Final Safety Analysis Report describes the offsite circuits as physically independent, therefore this is an administrative change. This change is consistent with NUREG-1432.

A.3 Current Technical Specification 3.8.1.1 does not contain any Actions when three or more required AC Sources are inoperable. Therefore if this Condition occurred, the CTS would require Specification 3.0.3 to be entered. Improved Technical Specification 3.8.1 will add an Action (Action I) which will require LCO 3.0.3 to be entered when three or more AC Sources are inoperable. Since there is no difference between the Actions that are required to be taken in CTS and ITS; this change is considered administrative. This change is consistent with NUREG-1432.

A.4 Improved Technical Specification 3.8.1 Action F (when one required offsite source is inoperable and one diesel generator [DG] is inoperable) contains a Note which requires the Distribution Systems - Operating Specification (LCO 3.8.9) Action to be entered when there is no AC power source to any train. This Note was added as a reminder to ensure that the appropriate Actions are entered when the 4.16 kV bus is de-energized. This Note takes exception to ITS LCO 3.0.6 (resulting in cascading in the ITS), which allows only eight hours to restore the de-energized bus. The CTS, through cascading, would also only allow eight hours to restore the de-energized bus. The Note does not impact any Completion Time; the time to restore the de-energized bus is the same in the CTS and ITS. Therefore, since the time to restore the de-energized bus remains the same, this change is considered administrative. This change is consistent with NUREG-1432.

A.5 Current Technical Specification 3.8.1.1 Actions c, d, and e (for two required offsite inoperable, one offsite and one DG inoperable, and two DGs inoperable, respectively) requires Actions to be taken once in these Conditions, which are the same as those required in CTS 3.8.1.1 Actions a and b (for one offsite circuit inoperable and one DG inoperable,

BASES

LCO (continued)

OPERABLE to support required trains of distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems—Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents). (1)

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies provide assurance that: (1) 3

a. Required features needed to mitigate a fuel handling accident are available;

b. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Channel

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4. (9)

ACTIONS

3

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1,2,3 or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1,2,3 or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. 9 (9)

The ACTIONS have been modified by a second Note stating that performance of REQUIRED ACTIONS shall not preclude completion of Actions to establish a safe conservative position. This clarification is provided to avoid stopping movement of irradiated fuel assemblies while in a non-conservative position based on compliance with the REQUIRED ACTIONS. 13

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required per LCO 3.8.10, the remaining train with DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to 3

(continued)

Page Replacement Instructions
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Section 3.9

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	**INSERT**
CTS Markup & Discussion of Changes	
Specification 3.9.4, Unit 1	Specification 3.9.4, Unit 1
Page 1 of 1	Page 1 of 1
Specification 3.9.4, Unit 2	Specification 3.9.4, Unit 2
Page 1 of 1	Page 1 of 1
DOC 3.9.4-1 through DOC 3.9.4-4	DOC 3.9.4-1 through DOC 3.9.4-4

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

Specification 3.9.4

3.9 ~~3/4.9~~ REFUELING OPERATIONS

3.9.4 ~~3/4.9.8~~ SHUTDOWN COOLING AND COOLANT CIRCULATION - High Water Level (A.1)

LIMITING CONDITION FOR OPERATION (M.3)

LCO 3.9.4 ~~3.9.8.1~~ ~~At least~~ one shutdown cooling loop shall be OPERABLE and in operation. (A.1)

APPLICABILITY: MODE 6 ~~at all reactor water levels~~. with ≥ 23 feet of water above the top of the irradiated fuel assemblies seated in the reactor vessel (A.6)

ACTION: loading irradiated fuel assemblies in the core and (A.5)

Required Action A.3 / Required Action A.2

a. With less than one shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and, specifically, the charging pumps shall be de-energized and the charging flow paths shall be closed. Close all containment penetrations providing direct access from one containment atmosphere to the outside atmosphere within 4 hours. (A.1) (A.5) (LA.1)

LCO NOTE 2

The shutdown cooling pumps may be de-energized during the time intervals required for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification ~~4.6.1.2.d~~ SR 3.6.1.1 ~~and/~~or to permit maintenance on valves located in the common shutdown cooling suction line, provided (1) no operations are permitted which could cause dilution of the Reactor Coolant System boron concentration and, specifically, the charging pumps shall be de-energized and the charging flow paths shall be closed, (2) all CORE ALTERATIONS are suspended, (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are maintained closed, and (4) the water level above the top of the irradiated fuel is greater than 23 feet. (A.1) (LA.1) (A.4)

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ (A.3)

A.1 Initiate action to restore SDC loop to OPERABLE status and operation. (M.1)

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 ~~4.9.8.1~~ A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 1500 gpm at least once per ~~4~~ 12 hours. (L.2)

LCO NOTE 1

The shutdown cooling loop may be not in ~~be removed from~~ operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. (A.1) (L.3)

provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration. (M.2)

Specification 3.9.4

1

4

3.9 3/4.9 REFUELING OPERATIONS

3.9.4 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION - High Water Level — A.1

LIMITING CONDITION FOR OPERATION — M.3 13

Operable and

LCO 3.9.4 ~~3.9.8.1~~ ~~At least~~ one shutdown cooling loop shall be in operation. ① — A.1

APPLICABILITY: MODE 6 ~~at all reactor water levels~~. With ≥ 23 ft of water above the top of the irradiated fuel assemblies seated in the reactor vessel — A.6

ACTION: loading irradiated fuel assemblies in the core and — A.5

Required Action A.3

Required Action A.2

a. With less ~~than one shutdown cooling loop in operation~~ ②, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System ~~and, specifically, the charging pumps shall be de-energized and the charging flow paths shall be closed.~~ Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. — A.5 — LA.1 — 4

LCO NOTE 2

The shutdown cooling pumps may be de-energized during the time intervals required for local leak rate testing of containment penetration number 41 pursuant to the requirements of Specification ~~4.6.1.2.c~~ and/or to permit maint[illegible] on valves located in the common shutdown cooling suctio[illegible] provided (1) no operations are permitted which could cause [illegible] of the Reactor Coolant System boron concentration ~~and, [illegible] the charging pumps shall be de-energized and the charg[illegible] flow paths shall be closed~~, (2) all CORE ALTERATIONS are suspe[illegible] (3) all containment penetrations providing direct access from [illegible] containment atmosphere to the outside atmosphere are maintained closed, and (4) the water level above the top of the irradiated fuel is greater than 23 feet. — SR 3.6.1.1 — A.1 — LA.1 — A.4 — 4

b. ~~The provisions of Specification 3.0.3 are not applicable.~~ — A.3

A.1 Initiate action to restore SDC loop to OPERABLE status and operation. — M.1

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 ~~4.9.8.1~~ A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 1500 gpm at least once per ~~4~~ 12 hours. — L.2 — 4 — 1

be not in — A.1 — 4

LCO NOTE 1 ① The shutdown cooling loop may ~~be removed from~~ operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS ~~in the vicinity of the reactor pressure vessel hot legs.~~ — L.3

provided no operations are permitted that would cause reduction of the Reactor Coolant System concentration. — M.2

ADMINISTRATIVE CHANGES

A.1 The proposed change will reformat, renumber, and reword the existing Technical Specifications with no change of intent to be consistent with NUREG-1432. As a result, the Technical Specifications should be more easily readable, and therefore, understandable by plant operators as well as other users.

During the Calvert Cliffs ITS development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe each LCO and to be consistent with NUREG-1432. However, the additional information does not change the intent of the current Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to existing Specifications.

A.2 Not used.

A.3 Current Technical Specification 3.9.8.1 Action states that the provisions of Specification 3.0.3 are not applicable. Improved Technical Specification 3.9.4 will not contain this requirement. Improved Technical Specification LCO 3.0.3 is only applicable in Modes 1, 2, 3, and 4. Therefore, LCO 3.0.3 is not applicable to this Mode 6 Specification. Removal of a requirement from a specific Technical Specification because it exists in Section 3.0 is considered administrative. This change is consistent with NUREG-1432.

A.4 The CTS LCO 3.9.8.1 Action a requirement that allows the SDC pumps to be de-energized contains a requirement that the water level above the top of the irradiated fuel assemblies be greater than 23 feet. Improved Technical Specification 3.9.4 LCO Note 2 contains the same allowance but does not contain the requirement that the water level be greater than 23 feet above the irradiated fuel assemblies seated in the reactor vessel. The requirement that the water level be greater than 23 feet above the irradiated fuel assemblies is not required because the Applicability contains the requirement. The deletion of a requirement which is part of the Applicability constitutes an administrative change. This change is consistent with NUREG-1432.

A.5 Current Technical Specifications LCO 3.9.8.1 Action a includes an action to suspend all operations involving an increase in the reactor decay heat load. Improved Technical Specification 3.9.4 Action A.2 is to suspend loading irradiated fuel assemblies in the core. Loading irradiated fuel assemblies in the core is essentially the only way to increase the reactor decay heat load. Since there are no differences between the Actions that are required to be taken in CTS and ITS, this change is considered administrative. This change is consistent with NUREG-1432.

A.6 Current Technical Specifications LCO 3.9.8.1 Applicability requires that one SDC loop be in operation when in Mode 6 at all reactor water levels. Improved Technical Specification 3.9.4 Applicability will require one SDC loop to be Operable and in operation in Mode 6 with ≥ 23 feet of water above the top of the irradiated fuel assemblies seated in the reactor vessel. This CTS was divided into two requirements in the ITS: one for water level ≥ 23 feet above the top of irradiated fuel assemblies, and one for water level < 23 feet above the top of the irradiated fuel assemblies. This change is for the case when water level is

≥ 23 feet above the top of irradiated fuel assemblies. Specification 3.9.5 is for the case when water level is < 23 feet above the top of the irradiated fuel assemblies. This change only addresses dividing CTS 3.9.8.1 into two specifications based on changes to the Applicability (ITS 3.9.4 and ITS 3.9.5). Since the CTS 3.9.8.1 Applicability of all reactor water levels includes both ≥ 23 feet of water above irradiated fuel assemblies seated in the reactor vessel and < 23 feet of water above irradiated fuel assemblies seated in the reactor vessel the change is considered to be administrative. Other changes to CTS 3.9.8.1 related to ITS 3.9.4 (water level ≥ 23 feet above irradiated fuel assemblies seated in the reactor vessel) are described in the Discussion of Changes for ITS 3.9.4. Other changes to CTS 3.9.8.1 related to ITS 3.9.5 (water level < 23 feet above irradiated fuel assemblies seated in the reactor vessel) are described in the Discussion of Changes for ITS 3.9.5. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Current Technical Specification 3.9.8.1 does not contain a requirement to immediately initiate action to restore one SDC loop to Operable status and operation. This requirement exists as an option to the other Actions (the option to restore always exists). Improved Technical Specification 3.9.4, Required Action A.1 will require the immediate initiation of action to restore one SDC loop to Operable status and operation (it is "AND'ed" to the other Required Actions). This requirement will ensure that Action is taken to restore forced flow and decay heat removal capability, and to prevent thermal and boron stratification in the core. The addition of Required Actions is a more restrictive change. This change does not adversely affect plant safety because it adds a requirement to immediately initiate action to restore forced flow. This change is consistent with NUREG-1432.

M.2 Current Technical Specification 3.9.8.1 Action a footnote * allows the SDC loop to be removed from operation for up to one hour per eight-hour period. Improved Technical Specification 3.9.4 LCO Note 1 also allows the SDC loop to be taken out-of-service for up to one hour per eight hours, but it also contains the caveat that no operations are permitted that would cause reduction of the Reactor Coolant System (RCS) boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. Therefore, the addition of this limitation is a more restrictive change because no operations will be permitted that could ensure a reduction of RCS boron concentration. This change does not adversely affect plant safety because it requires additional limitations to prevent uneven boron distribution in the core. Removal of the * footnote limitation of only applying to Core Alterations is addressed in Discussion of Change L.3. This change is consistent with NUREG-1432.

M.3 Current Technical Specification LCO 3.9.8.1 requires one shutdown cooling (SDC) loop to be in operation. Improved Technical Specification 3.9.4 will require one SDC loop to be Operable and in operation. This requirement is implied because if the system is in operation and not complying with its SR, (which is the same for both CTS and ITS) it is declared inoperable. Adding a requirement that is implied is a more restrictive change. This change does not adversely affect plant safety because it adds a requirement that the operating SDC loop also be Operable. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - RELOCATIONS

None

TECHNICAL CHANGES - MOVEMENT OF INFORMATION TO LICENSEE-CONTROLLED DOCUMENTS

LA.1 Current Technical Specification 3.9.8.1 Action a requires that all operations which could involve a reduction in RCS boron concentration be suspended and, specifically to de-energize charging pumps and to close charging flow paths. Improved Technical Specification SR 3.9.4 for the same condition will require that operations that could involve a reduction in RCS boron concentration be suspended, but will not contain the details of the method ensure compliance used to with these requirements is maintained (i.e., de-energize the charging pumps and isolate charging flow paths). The requirements will be moved to the ITS Bases. This is acceptable because this detail is not required to meet the overall requirement since ITS 3.9.4, Required Action A.2, continues to require immediate suspension of operations involving a reduction in boron concentration. Changes to the Bases will be controlled by the Bases Control Program of ITS Chapter 5.0. This approach provides an effective level of control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change. Furthermore, NRC and Calvert Cliffs resources associated with processing license amendments to these requirements will be reduced. This is a less restrictive movement of information change with no effect on safety. This change is consistent with NUREG-1432.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 Not used.

L.2 Current Technical Specification 4.9.8.1 requires reactor coolant flow rate be determined ≥ 1500 gpm at least once per four hours. Improved Technical Specification SR 3.9.4.1 requires flow rate be determined ≥ 1500 gpm at least once per 12 hours. This change decreases the Frequency to verify SDC flow from four hours to twelve hours. The 12-hour Frequency is sufficient considering the flow, temperature, pump control, and alarm indications available to the operator in the Control Room for monitoring the SDC System. Also, as long as LCO 3.9.4.1 is met, which requires one SDC loop to be in operation, flow is not expected to change significantly with time. This verification is oriented toward ensuring no significant system line-up problem or system failure prevents flow from meeting requirements. Plant history indicates such problems are rare. Reducing the Surveillance Frequency constitutes a less restrictive change. This change is consistent with NUREG-1432.

L.3 Current Technical Specification 3.9.8.1 LCO and Action a footnote * allows the SDC loop to be removed from operation for up to one hour per eight-hour period during the performance of Core Alterations in the vicinity of the reactor pressure vessel hot legs. Improved Technical Specification LCO 3.9.4 Note allows the SDC loop to be removed from operation for up to one hour per eight-hour period any time in the Applicable Mode, provided no operations are permitted that would cause reduction of the RCS boron concentration. The allowance is no longer limited to only when Core Alterations are taking place in the vicinity of the hot legs. Therefore, SDC may be removed from operation to perform other activities such as core mapping, RCS to SDC isolation valve testing, etc. When SDC is in operation, flow from the reactor pressure vessel hot legs causes turbulence. When fuel is being moved in the vicinity of the hot legs, this causes the fuel to sway and the water is too turbulent to allow operators to load fuel. The concern when SDC is removed from operation for this short period of time is not decay heat removal but potential reactivity events, which can occur due to boron reductions. Since the boron concentration is not permitted to be reduced when in this condition, uniform concentration distribution is maintained and the reactivity will remain within limits. Removing this limit on when the Note is applicable constitutes a less restrictive change because it allows SDC to be removed from operation for any reason, provided no reductions in boron concentrations occur. This change is consistent with NUREG-1432.

Page Replacement Instructions
VOLUME 15
Section 5.0

Note: Underlined titles indicate tabs in volumes. Regarding CTS markups: Pages are referenced by citing the unit number as well as the specification number located in the upper right-hand corner of the CTS page.

Key:
DOC = Discussion Of Changes
DOD = Discussion Of Technical Specification Deviation or Discussion Of Bases Deviation

REMOVE	**INSERT**
ITS	
5.0-13 through 5.0-39	*5.0-13 through 5.0-40*
CTS Markup & Discussion of Changes	
DOC 5.0-1	DOC 5.0-1

Note: Italicized entries indicate uneven exchanges. Please follow page replacement instructions carefully.

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a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.9 Steam Generator Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program except as specified for individual requirements. This program provides controls for the inservice inspection of steam generator tubes to ensure that structural integrity of this portion of the Reactor Coolant System is

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maintained. The program shall contain the requirements listed below.

a. <u>Steam Generator Sample Selection and Inspection</u> - The minimum number of steam generators to be inspected shall be determined as specified in Table 5.5.9-1.

b. <u>Steam Generator Tube Sample Selection and Inspection</u> - The
steam generator tube minimum sample size, inspection result
classification, and the corresponding action required shall
be as specified in Tables 5.5.9-2 and 5.5.9-3. The inservice | 13
inspection of steam generator tubes shall be performed at the
Frequencies specified in Specification 5.5.9.c and the | 11
inspected tubes shall be verified acceptable per the
acceptance criteria of Specification 5.5.9.d. When applying
the exceptions of 5.5.9.b.1 through 5.5.9.b.3, previous
defects or imperfections in the area repaired by sleeving are
not considered an area requiring reinspection. The tubes
selected for each inservice inspection shall include at least
3% of the total number of tubes in all steam generators; the
tubes selected for these inspections shall be selected on a
random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

2. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

    i. All nonplugged tubes that previously had detectable
    wall penetrations (> 20%); and | 11

    ii. Tubes in those areas where experience has indicated potential problems.

3. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the

inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the three categories specified below. In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the percentage calculations.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected, are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes, or more than 1% of the inspected tubes are defective.

c. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following Frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months, but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If at least 20 percent of the tubes were inspected and the results were in the C-1 Category, or if at least 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended

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up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in the C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections following service under all-volatile treatment 11
conditions, not including the preservice inspection result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the inservice inspection results of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 13
at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1; the interval may then be extended to a maximum of once per 30 or 40 months, as applicable.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and 13

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5.5.9-3 during the shutdown subsequent to any of the following conditions: |13

i. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; |11

ii. A seismic occurrence greater than the Operating Basis Earthquake; |11

iii. A loss-of-coolant accident requiring actuation of the engineered safeguards; or |11

iv. A main steam line or feedwater line break.

4. The provisions of Specification SR 3.0.2 do not apply for extending the Frequency for performing inservice inspections as stated in Specifications 5.5.9.c.1 and 5.5.9.c.2.

d. Acceptance Criteria - As used in this Specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections. |11

3. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

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5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

i.	original tube wall	40%
ii.	Westinghouse laser welded sleeve wall	40%
iii.	ABB-Combustion Engineering leak tight sleeve wall	28%

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8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3 above.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

    i. Westinghouse Laser Welded Sleeving as described in the proprietary Westinghouse Reports WCAP-13698,

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Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, Generic Sleeving Report," April 1995; and WCAP-14469, "Specific Application of Laser Welded Sleeving for the Calvert Cliffs Power Plant Steam Generators," November 1995.

ii. ABB-Combusting Engineering Leak Tight Sleeving as described in the proprietary ABB-Combustion Engineering Report CEN-630-P, Revision 01, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," August 1996. A post-weld heat treatment during installation will be performed. 13

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 5.5.9.d.9 is required prior to returning previously plugged tubes to service.

e. Surveillance Completion - The Steam Generator Tube Surveillance Program is met after completing the corresponding actions (plug or repair all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 5.5.9-2 and 5.5.9-3. 13

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Table 5.5.9-1
Minimum Number of Steam Generators to be Inspected During Inservice Inspection

Preservice Inspection	No			Yes		
No. Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequence shall be modified to inspect the most severe conditions.

[2] The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

[3] Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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Table 5.5.9-2
Steam Generator Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per steam generator	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this steam generators	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this steam generator.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this steam generator, plug or repair defective tubes and inspect 2S tubes in each other steam generator. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	All other steam generators are C-1	None	N/A	N/A
			Same steam generators C-2 but no additional steam generator are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional steam generator is C-3	Inspect all tubes in each steam generator and plug or repair defective tubes. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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Table 5.5.9-3
Steam Generator Repaired Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A Minimum of 20% of repaired tubes[1][2]	C-1	None	N/A	N/A
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this SG.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this SG, plug defective tubes and inspect 20% of the repaired tubes in the other SG. 24-Hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4	Other SG is C-1	None
			Other SG is C-2	Perform action for C-2 result of first sample
			Other SG is C-3	Inspect all repaired tubes in each SG and plug defective tubes. 24-hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from either SG based on outage plans.

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5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which are required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of engineered safety feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.11.a and 5.5.11.b shall be performed once per 18 months for ventilation systems other than the Iodine Removal System (IRS) and 24 months for the IRS; after each complete or partial replacement of the high efficiency particulate air (HEPA) filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system. 2

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Tests described in Specification 5.5.11.c shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS; after 720 hours of system operation; after any structural maintenance on the HEPA filter or charcoal adsorber housing; and following painting, fire, or chemical release in any ventilation zone communicating with the system.

Tests described in Specification 5.5.11.d shall be performed once per 18 months for ventilation systems other than the IRS and 24 months for the IRS.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program test frequencies.

a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System (CREVS)	2,000 cfm
Emergency Core Cooling System (ECCS) Pump Room Exhaust Filtration System (PREFS)	3,000 cfm
Penetration Room Exhaust Ventilation System (PREVS)	2,000 cfm
Spent Fuel Pool Exhaust Ventilation System (SFPEVS)	32,000 cfm
IRS	20,000 cfm

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52,

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Revision 2, and ANSI N510-1975, at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Flowrate
CREVS	2,000 cfm
ECCS PREFS	3,000 cfm
PREVS	2,000 cfm
SFP Ventilation System	32,000 cfm
IRS	20,000 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide (elemental iodine for the IRS) penetration less than or equal to the value specified below when tested in accordance with ANSI N510-1975 and the testing protocol of ANSI D3803-89 at a temperature of ≤ 30°C (130°C for the IRS) and greater than or equal to the relative humidity specified as follows: 11 2 12

ESF Ventilation System	Penetrations	RH
CREVS	10%	95%
ECCS PREFS	10%	95%
PREVS	10%	95%
SFP Ventilation System	10%	95%
IRS	5%	95%

d. For each of the ESF systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52,

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Revision 2, and ANSI N510-1975 at the system flowrate specified as follows ± 10%:

ESF Ventilation System	Delta P	Flowrate
CREVS	4 inwg	2,000 cfm
ECCS PREFS	4 inwg	3,000 cfm
PREVS	6 inwg	2,000 cfm
SFP Ventilation System	4 inwg	32,000 cfm
IRS	6 inwg	20,000 cfm

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5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides control for potentially explosive gas mixtures contained in the Waste Gas Holdup System and the quantity of radioactivity contained in gas storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in the ODCM.

The program shall include:

a. The limits for concentrations of oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and 11

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 58,500 curies noble gases (considered as Xe-133).

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance Frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A Diesel Fuel Oil Testing Program to implement required testing of both new fuel oil and stored fuel oil shall be established. The

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program shall include sampling and testing requirements, and acceptance criteria, all in accordance with ASTM Standards. The purpose of the program is to establish the following: 11

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. An American Petroleum Institute gravity or an absolute specific gravity within limits, 11

2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. Water and sediment ≤ 0.05%.

b. Within 31 days following addition of new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil, when determined by gravimetric analysis based on ASTM D2276-1989, is ≤ 10 mg/l when tested every 92 days. 2 11

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Frequencies. 11

5.5.14 Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the Technical Specifications incorporated in the license; or

2. A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 <u>Safety Function Determination Program (SFDP)</u> 11

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into Limiting Condition for Operation (LCO) 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following: 11

a. Provisions for cross-train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

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A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a 11
loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, 7
Appendix J, Option B. This program shall be in accordance with
the guidelines contained in Regulatory Guide 1.163, "Performance- 2
Based Containment Leak-Test Program," dated September 1995, 7
including errata.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, is 49.4 psig. The containment design pressure is 50 psig.

5.5 Programs and Manuals

The maximum allowable containment leakage rate, L_a, shall be 0.20 percent of containment air weight per day at P_a. 7

Leakage rate acceptance criteria are: 7

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing, in accordance with this program, the leakage rate acceptance criterion are $\leq 0.60\ L_a$ for Types B and C tests and $\leq 0.75\ L_a$ for Type A tests. 7

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.0002\ L_a$ when pressurized to ≥ 15 psig. 11

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program. 2

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

---------------------------------NOTE---------------------------------

A single submittal may be made for both units, but shall not include the occupational radiation exposure from the Independent Spent Fuel Storage Installation. The submittal should combine sections common to both units at the station. 2

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). 11 This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, electronic personal dosimeter, or thermoluminescent dosimeter. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

---------------------------------NOTE---------------------------------

A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the

5.6 Reporting Requirements

reporting period. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10 CFR Part 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The report shall identify the thermoluminescent dosimeter results that represent collocated dosimeters in relation to the NRC thermoluminescent dosimeter program, and the exposure period associated with each result. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a, as modified by approved exemptions. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM, Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

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5.6 Reporting Requirements

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	SHUTDOWN MARGIN
3.1.3	Moderator Temperature Coefficient
3.1.6	Regulating Control Element Assembly Insertion Limit
3.2.1	Linear Heat Rate
3.2.2	Total Planar Radial Peaking Factor
3.2.3	Total Integrated Radial Peaking Factor
3.2.5	AXIAL SHAPE INDEX
3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. CENPD-199-P, Latest Approved Revision, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986

2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II," December 1979

3. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," January 1980

5.6 Reporting Requirements

4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2," March 1980

5. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

6. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

7. CEN-348(B)-P, "Extended Statistical Combination of Uncertainties," January 1987

8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

9. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986

10. CENPD-162-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

11. CENPD-207-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

12. CENPD-206-P-A, Latest Approved Revision, "TORC Code, Verification and Simplified Modeling Methods"

5.6 Reporting Requirements

13. CENPD-225-P-A, Latest Approved Revision, "Fuel and Poison Rod Bowing"

14. CENPD-266-P-A, Latest Approved Revision, "The ROCS and DIT Computer Code for Nuclear Design"

15. CENPD-275-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

16. CENPD-382-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Erbium Burnable Absorbers"

17. CENPD-139-P-A, Latest Approved Revision, "C-E Fuel Evaluation Model Topical Report"

18. CEN-161-(B)-P-A, Latest Approved Revision, "Improvements to Fuel Evaluation Model"

19. CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1989

20. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

21. CEN-372-P-A, Latest Approved Revision, "Fuel Rod Maximum Allowable Gas Pressure"

22. Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

23. CENPD-132, Supplement 3-P-A, Latest Approved Revision, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"

5.6 Reporting Requirements

24. CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985

25. CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985

26. Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"

27. CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977

28. Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

29. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977

30. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977

31. Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

32. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977

33. Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

5.6 Reporting Requirements

34. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

35. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

36. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

37. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

38. Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

39. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

40. CENPD-188-A, Latest Approved Revision, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

41. The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:

    i. CENPD-153-P, Latest Approved Revision, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

5.6 Reporting Requirements

ii. CEN-199(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2," November 1979

iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements

5.6.6 Pressurizer Power-Operated Relief Valve and Safety Valve Report

A report shall be submitted prior to March 1 of each year documenting all failures of and challenges to the pressurizer power-operated relief valves, or safety valves.

5.6.7 Post-Accident Monitoring Report

When a report is required by Condition B or G of LCO 3.3.10, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. 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 of the Function to OPERABLE status.

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.9 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 15 days.

b. The complete results of the steam generator tube inservice inspection during the report period shall be submitted to the NRC prior to March 1 of each year. This report shall include:

   1. Number and extent of tubes inspected;

5.6 Reporting Requirements

2. Location and percent of wall-thickness penetration for each indication of an imperfection; and | 11

3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days.

ADMINISTRATIVE CHANGES

A.1 The proposed changes reformat, renumber, and reword the Current Technical Specification (CTS) definitions, with no change of intent, in accordance with NUREG-1432. As a result, the Technical Specifications should be more readily readable and, therefore, understandable by plant operators, as well as other users.

During the Calvert Cliffs Improved Technical Specifications (ITS) development, certain wording preferences or conventions were adopted which resulted in no technical changes to the Technical Specifications. Additional information may also have been added to more fully describe each section and to be consistent with NUREG-1432. However, this additional information does not change the intent of the CTS. The reformatting, renumbering, and rewording process involves no technical changes to existing Definitions.

A.2 Current Technical Specification 6.2.2.a requires at least three non-licensed Operators to be assigned to shift crews and licensed Operators to be present in the Control Room during specific plant operation. Improved Technical Specifications 5.2.2.a and 5.2.2.b will delete the words, "at least," from the requirement. The convention of the ITS is to list the minimum requirement which is always allowed to be exceeded (the plant is always allowed to be more conservative, i.e., having more than three non-licensed operators). Since this change does not revise the minimum requirement, this change is considered administrative in nature. This change is consistent with NUREG-1432.

A.3 Current Technical Specification 6.2.2.d requires an individual qualified in radiation protection procedures to be onsite when fuel is in the reactor. Improved Technical Specification 5.2.2.e changes the title of the individual to a radiation protection technician. This change reflects the generic title of the individual specified in the American National Standards Institute standards. This change is consistent with NUREG-1432, Generic Change TSTF-65.

A.4 Current Technical Specification 6.3.1, which requires the Shift Technical Advisor to have specific education requirements, is being incorporated into the Shift Technical Advisor requirements of CTS 6.2.2.g.2. This change is considered administrative because the requirements have not changed. Current Technical Specification 6.2.2.g.2 references CTS 6.3.1. This change will delete the reference to another Specification and incorporate the actual words. The ITS number for this requirement is ITS 5.2.2.g.

A.5 Current Technical Specification 6.2.1.a requires lines of authority, responsibility, and communication to be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. It further requires that these relationships be documented in organizational charts, functional descriptions . . . and the Updated Final Safety Analysis Report (UFSAR). Improved Technical Specification 5.2.1.a is consistent with the CTS except it also allows the plant specific titles of personnel fulfilling the responsibilities of the position delineated in the Technical Specifications to be placed in the Quality Assurance (QA) Policy, instead of or along with the UFSAR. This change is administrative in nature because these requirements will be adequately controlled in either the UFSAR or QA Policy. This change is consistent with NUREG-1432, Generic Change TSTF-65.