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Constellation Nuclear

Calvert Cliffs Nuclear Power Plant

A Member of the Constellation Energy Group

December 20, 2000

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request: Revision to the Technical Specifications to Support Steam Generator Replacement

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) hereby requests an amendment to Renewed Operating License Nos. DPR-53 and DPR-69 to incorporate the changes described below into the Technical Specifications for CCNPP Units 1 and 2.

DESCRIPTION

Calvert Cliffs Nuclear Power Plant, Inc. will be replacing the Combustion Engineering Model 67 Steam Generators with steam generators fabricated by Babcock & Wilcox Canada Ltd. (BWC) during the Unit 1 and Unit 2 refueling and replacement outages in the spring of 2002 and 2003, respectively. The proposed amendment would revise the CCNPP Technical Specifications to incorporate changes required to support operation with the replacement steam generators. The proposed changes are: (1) incorporating a new reference point for the "reactor trip steam generator level-low" set-point to account for geometric difference between the replacement steam generators and original steam generators; (2) increasing reactor coolant minimum required total flow rate back to the originally established value of 370,000 gpm from the current value of 340,000 gpm, which was recently established to accommodate more tube plugging in the original steam generators; and (3) deleting the steam generator tube sleeving options previously approved by the Nuclear Regulatory Commission for the original steam generators. The detailed descriptions and the safety analyses for the proposed revisions are provided in Attachment (1).

REQUESTED CHANGE

Change the CCNPP Technical Specification Table 3.3.1-1, Limiting Condition for Operation 3.4.1, Surveillance Requirement 3.4.1.3, and Administrative Control 5.5.9 as shown on the marked-up pages in Attachment (3).

A001

ASSESSMENT AND REVIEW

We have evaluated the significant hazards considerations associated with this proposed change, as required by 10 CFR 50.92, and have determined that there are none (see Attachment 2 for a complete discussion). We have also determined that operation with the proposed modification will not result in any significant change in the types or significant increases in the amounts of any effluents that may be released offsite, and no significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed amendment.

SAFETY COMMITTEE REVIEW

The proposed amendment to the CCNPP Technical Specifications has been reviewed by our Plant Operations and Safety Review Committee and Offsite Safety Review Committee. They have concluded that implementing this amendment will not result in an undue risk to the health and safety of the public.

SCHEDULE

The proposed amendment is scheduled to be implemented following the refueling and replacement outages in spring 2002 for Unit 1 and spring 2003 for Unit 2. To help us prepare for the replacement outage in a timely fashion, we request that you review and approve our application by June 30, 2001, for delayed implementation.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

STATE OF MARYLAND :
: TO WIT:
COUNTY OF CALVERT :

I, Charles H. Cruse, being duly sworn, state that I am Vice President, Nuclear Energy, Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 20th day of December, 2000.

WITNESS my Hand and Notarial Seal:

MELISSA DAWN RUSK
NOTARY PUBLIC STATE OF MARYLAND
My Commission Expires November 1, 2003

My Commission Expires: 11-01-03

12/20/00
Date

CHC/GT/dlm

Attachments: (1) Proposed Revision to CCNPP Technical Specifications to Support Steam Generator Replacement; Description and Safety Evaluation
(2) Determination of Significant Hazards
(3) Marked-Up Technical Specification Pages

cc: R. S. Fleishman, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

PROPOSED REVISION TO CCNPP TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

Calvert Cliffs Nuclear Power Plant, Inc.
December 20, 2000

ATTACHMENT (1)

PROPOSED REVISION TO CCNPP TECHNICAL SPECIFICATIONS TO SUPPORT STEAM GENERATOR REPLACEMENT; DESCRIPTION AND SAFETY EVALUATION

I. BACKGROUND

Calvert Cliffs Nuclear Power Plant (CCNPP) is a dual unit site. Each unit is a two-loop 2700 MWt Combustion Engineering (CE) design. The original CE Model 67 Steam Generators have been in service since the mid-seventies when the two CCNPP units were first licensed for commercial operation. Calvert Cliffs is currently preparing to replace CCNPP steam generators with steam generators fabricated by Babcock & Wilcox Canada Ltd. (BWC) during the Unit 1 refueling outage in the spring of 2002 (end of Cycle 15), and the Unit 2 refueling outage in the spring of 2003 (end of Cycle 14).

The replacement steam generator (RSG) consists of a new lower subassembly, new steam drum internals, new feedring, with the existing steam drum being refurbished and reattached to the subassembly within the containment. The RSGs will occupy the same physical envelope as the original steam generators (OSGs). There are no changes to interfaces with the reactor coolant, main feedwater, or main steam systems, or to major component supports or piping supports. Some of the differences between the OSG and RSG designs include: (1) use of thermally-treated Alloy 690 tube material instead of high temperature mill-annealed Alloy 600 used for the OSG; (2) a small weight increase and a small change in the center of gravity location; (3) addition of an integral flow restrictor in the main steam nozzle; (4) increased heat transfer area; and (5) a minor geometric difference in the location of the top of the feed ring with respect to the pedestal.

Steam generator replacement activity in the nuclear utility industry is now a fairly mature process that is handled routinely using the 10 CFR 50.59 process. Accordingly, the Calvert Cliffs steam generator replacement is also being conducted using the 50.59 process. We do not anticipate any issues requiring Nuclear Regulatory Commission (NRC) prior approval as a result of the 10 CFR 50.59 evaluation. This application revises the CCNPP Technical Specifications to incorporate changes required:

(1) By the change to the location of feed ring;

(2) To re-establish reactor coolant flow rate to original values; and

(3) To remove repair options that are not compatible with the RSGs.

II. DESCRIPTION OF PROPOSED CHANGE

A. Technical Specification Table 3.3.1-1, Item 7

Technical Specification Table 3.3.1-1, "Reactor Protective System Instrumentation," Item 7, sets the allowable value for "steam generator level-low" function to greater than or equal to 10 inches below the top of the feed ring. We are proposing to change the allowable value to greater than or equal to 50 inches below normal water level.

The reason for this change is to accommodate the geometric difference in the location of the top of the feed ring with respect to the pedestal between the OSG (510.8 inches) and the RSG (484.8 inches), and to introduce "normal water level" as a more practical reference point for Reactor Protective System steam generator level-low set point. Normal water level is a more practical and appropriate frame of reference since it is the frame of reference for steam generator water level indication used in the Control Room by the operators.

The normal water levels for RSG and OSG with respect to the pedestal are identical. The current steam generator level-low reactor trip setpoint, "≥ 10 inches below top of feed ring" is

"≥ 50 inches below normal water level" for both the RSG and OSG. As illustrated in the comparison table below, the percent narrow range span and normal water level in percent span are identical between the RSGs and OSGs. The functionality of the steam generator level-low reactor trip setpoint will, therefore, be unchanged.

We are proposing a revision to the CCNPP Technical Specification Table 3.3.1-1 as shown on the marked-up pages in Attachment (3).

RSG-OSG Comparison Table

(Dimensions Are Measured In Inches From The Steam Generator Pedestal)

Parameter	OSG	RSG
Narrow Range Water Level Lower Instrument Tap	434.4"	434.4"
Narrow Range Water Level Upper Instrument Tap	614.3"	614.3"
Narrow Range Water Level Instrument Span	614.3 – 434.4 = 179.9"	614.3 – 434.4 = 179.9"
Normal Water Level	550.8"	550.8"
Normal Water Level as a percent of NR Span	(550.8 – 434.4)/179.9 = 64.7%	(550.8 – 434.4)/179.9 = 64.7%
Top of Feed Ring Location	510.8"	484.8"
Top of Feed Ring Location relative to Normal Water Level	550.8 – 510.8 = 40 inches below Normal Water Level	N/A
Steam Generator Level-Low Rx Trip Setpoint	10 inches below Top of Feed Ring: 510.8 – 10 = 500.8" Note: Setpoint is 50 inches below Normal Water Level: 550.8 – 50 = 500.8"	50 inches below Normal Water Level: 550.8 – 50 = 500.8"
Steam Generator Level-Low Rx Trip Setpoint as a percent of NR Span	(500.8 – 434.4)/179.9 = 36.9%	(500.8 – 434.4)/179.9 = 36.9%
Secondary Side Water Inventory Mass at 116.4 inches below Normal Water Level	64,049 lbm	64,115 lbm

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

By letter dated May 23, 1998 (Reference 1) the NRC issued an amendment revising the Calvert Cliffs Technical Specifications to decrease the required minimum Reactor Coolant System

(RCS) total flow rate from 370,000 gpm to 340,000 gpm to support increased steam generator tube plugging. The flow resistance of the RSGs is equivalent to that of the OSGs with zero plugged tubes. Therefore, the required minimum RCS total flow rate can be increased to the value previously established for the OSGs with zero plugged tubes, 370,000 gpm.

Since the RCS total flow rate will increase to that previously associated with the OSGs with zero plugged tubes, we are proposing an increase to the required minimum RCS total flow rate from 340,000 gpm to 370,000 gpm in order to provide an opportunity to credit more RCS flow in future revisions to existing safety analyses. Crediting more RCS flow in the safety analyses allows for greater flexibility in core design and operation.

We are proposing a revision to the CCNPP LCO 3.4.1 and Surveillance Requirement 3.4.1.3 as shown on the marked-up pages in Attachment (3).

C. Technical Specification Administrative Control 5.5.9

As described above in the background section, one of the differences between the OSG and the RSG design is the use of thermally-treated Alloy 690 tube material instead of high temperature mill-annealed Alloy 600 used for the OSG. The three sleeving tube repair options described in CCNPP Technical Specification Administrative Control 5.5.9, the Westinghouse Laser Welded sleeves (Reference 2), the Asea Brown Boveri, Inc. (ABB)-Combustion Engineering Leak Tight sleeves (Reference 3), and the ABB-Combustion Engineering Alloy 800 Leak Limiting sleeves (Reference 4), are designed specifically for the OSGs' mill-annealed Alloy 600 tubes. As currently designed and described in Administrative Control 5.5.9, these sleeving options are not compatible and cannot be used to repair the RSG Alloy 690 tubes. Therefore, these repair options must be deleted from Administrative Control 5.5.9.

We are proposing a revision to the CCNPP Technical Specification Administrative Control 5.5.9 as shown on the marked-up pages in Attachment (3).

III. SAFETY ANALYSIS

A. Technical Specification Table 3.3.1-1, Item 7

The Technical Specification Basis for the allowable value for reactor trip on steam generator low level is for this setpoint to be sufficiently below the normal operating water level so as not to cause a reactor trip during normal operations, and high enough to ensure that a reactor trip signal is generated to prevent operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The safety analysis for the Loss of Feedwater Flow design basis event assumes steam generator low water level trip at 116.4 inches below normal water level. This assumption provides margin for the Technical Specification Reactor Trip Steam Generator Level-Low Setpoint of 50 inches below normal water level. This assumption remains unchanged between the RSGs and the OSGs.

The Steam Generator Level-Low Reactor Trip Setpoint, in combination with the Auxiliary Feedwater Actuation System, ensures that adequate secondary side water inventory exists in both RSGs to remove decay heat following a Loss of Feedwater Flow event. The acceptance criteria for the Loss of Feedwater Flow event are: (1) peak RCS pressure shall not exceed the RCS pressure upset limit. (2) steam generator dryout does not occur. These criteria assure that fuel cladding integrity is maintained by ensuring that departure from nucleate boiling ratio (DNBR) meets the DNBR criterion. To ensure that the acceptance criteria will be met with the RSGs, there must be as much mass in RSG at the Safety Analysis water level than OSG. The OSG Safety Analysis water level is 116.4 inches below normal water level. Using the same method to predict steam generator inventory, at this water level, OSG has 64,049 lbm water mass and RSG has 64,115 lbm water mass. Therefore, the RSG has more post-reactor trip secondary side inventory than the OSG to ensure the Loss of Feedwater Flow event acceptance criteria are not challenged.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

Increasing the required minimum RCS total flow rate has no adverse impact on the plant's safety analyses. The increase in RCS flow associated with the replacement steam generators is within the bounds previously analyzed for the original steam generators. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal. Credit for this enhanced core heat removal may then be taken in the safety analyses. The overall core thermal margin in the safety analyses will remain essentially the same.

C. Technical Specification Administrative Control 5.5.9

The three sleeving options described in Administrative Control 5.5.9 were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Administrative Control 5.5.9 has no safety significance.

IV. REFERENCES

1. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated May 23, 1998, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit 1 (TAC No. M97855) and Unit 2 (TAC No. M97856)
2. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. C. H. Cruse (BGE), dated March 22, 1996, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit 1 (TAC No. M94205) and Unit 2 (TAC No. M94206)
3. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated November 18, 1997, Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit 1 (TAC No. M94205) and Unit 2 (TAC No. M94206)
4. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated September 1, 1999, Issuance of Amendment Re: Steam Generator Tube Repair Using Leak Limiting Alloy 800 Sleeves Unit 1 (TAC No. MA4278) and Unit 2 (TAC No. MA4279)

DETERMINATION OF SIGNIFICANT HAZARDS

Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP) will be replacing the Combustion Engineering Model 67 original steam generators with replacement steam generators fabricated by Babcock & Wilcox Canada Ltd. during the Unit 1 and Unit 2 refueling and replacement outages in the spring of 2002 and 2003, respectively. The proposed amendment would revise the CCNPP Technical Specifications to incorporate changes required: (1) by the change to the location of feed ring; (2) to re-establish reactor coolant flow rate to original values; and (3) to remove repair options that are not compatible with the replacement steam generators.

The proposed changes have been evaluated against the standards in 10 CFR 50.92 and have been determined to not involve a significant hazards consideration in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   A. Technical Specification Table 3.3.1-1, Item 7

   Technical Specification Table 3.3.1-1, "Reactor Protective System Instrumentation," Item 7 sets the allowable value for "Steam Generator Level-Low" function to greater than or equal to 10 inches below the top of the feed ring. To accommodate the geometric difference in the location of the top of the feed ring with respect to the pedestal between the original steam generators (OSG) (510.8 inches) and the replacement steam generators (RSG) (484.8 inches), the proposed amendment would change the allowable value for "Steam Generator Level-Low" function to greater than or equal to 50 inches below normal water level. Since normal water levels for RSG and OSG with respect to the pedestal are identical and the current steam generator level-low reactor trip setpoint, "$\geq$ 10 inches below top of feed ring" is "$\geq$ 50 inches below normal water level" for both the RSG and OSG, the functionality of the steam generator level-low reactor trip setpoint will be unchanged. Furthermore, use of normal water level as the point of reference instead of top of the feed ring is more practical and appropriate since it is the frame of reference for steam generator water level indication used in the Control Room by the operators.

   The design basis accident affected by the proposed change is the Loss of Feedwater Flow event. The Steam Generator Level-Low Reactor Trip Setpoint, in combination with the Auxiliary Feedwater Actuation System, ensures that adequate secondary side water inventory exists in both RSGs to remove decay heat following a Loss of Feedwater Flow event. To ensure that the acceptance criteria for the Loss of Feedwater Flow event are met with the RSGs, there must be at least as much mass in RSG at the Safety Analysis water level as in the OSG. The OSG Safety Analysis water level is 116.4 inches below normal water level. Using the same method to predict steam generator inventory, at this water level, OSG has 64,049 lbm water mass and RSG has 64,115 lbm water mass. Therefore, the RSG has more post-reactor trip secondary side inventory than the OSG which ensures the Loss of Feedwater event acceptance criteria are not challenged.

   Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function will not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

The proposed amendment would revise Technical Specification LCO 3.4.1 and Surveillance Requirement 3.4.1.3 to increase reactor coolant minimum required total flow rate back to the originally established value of 370,000 gpm from the current value of 340,000 gpm, which was recently established to accommodate more tube plugging in the OSG. The flow resistance of the RSG is equivalent to that of the OSG with zero plugged tubes. Therefore, the required minimum RCS total flow rate can be increased to the value previously established for the original steam generators with zero plugged tubes, 370,000 gpm.

Increasing the required minimum RCS total flow rate has no adverse impact on the safety analysis. Crediting more RCS flow in the safety analysis allows for greater flexibility in core design and operation. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal. The overall core thermal margin in the safety analysis will remain essentially the same.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate will not involve a significant increase in the probability or consequences of an accident previously evaluated.

C. Technical Specification Administrative Control 5.5.9

The proposed revision deletes three sleeving options from Administrative Technical Specification 5.5.9. The sleeving options are: Westinghouse Laser Welded sleeves, Asea Brown Boveri, Inc. (ABB)-Combustion Engineering Leak Tight sleeves, and the ABB-Combustion Engineering Alloy 800 Leak Limiting sleeves. One of the differences between the OSG and the RSG design is the use of thermally-treated Alloy 690 tube material instead of high temperature mill-annealed Alloy 600 used for the OSG. The three sleeving tube repair options described in Calvert Cliffs Nuclear Power Plant (CCNPP) Technical Specification Administrative Control 5.5.9, are designed specifically for the OSGs' mill-annealed Alloy 600 tubes.

The three sleeving options were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Administrative Control 5.5.9 has no safety significance.

Therefore, the proposed revision will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Would not create the possibility of a new or different type of accident from any accident previously evaluated.*

A. Technical Specification Table 3.3.1-1, Item 7

The RSGs are equivalent in function to the OSGs. Changing Technical Specification Table 3.3.1-1, Item 7 is required to provide a correct and practical reference point from which to measure the Reactor Trip Steam Generator Level-Low Setpoint. As described above in Item 1, the normal water levels for RSG and OSG with respect to the pedestal are identical and the current steam generator level-low reactor trip setpoint, "≥ 10 inches below top of feed ring" is

"≥ 50 inches below normal water level" for both the RSG and OSG. Hence, the functionality of the reactor trip steam generator level-low setpoint will be unchanged. Furthermore, use of normal water level as the point of reference instead of top of the feed ring is more practical and appropriate since it is the frame of reference for steam generator water level indication used in the Control Room by the operators.

Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function will not create the possibility of a new or different type of accident from any accident previously evaluated.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

As described above in Item 1, increasing the required minimum RCS total flow rate has no adverse impact on the plant's safety analyses. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate will not create the possibility of a new or different type of accident from any accident previously evaluated.

C. Technical Specification Administrative Control 5.5.9

As described in Item 1 above, the three sleeving options were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Technical Specification Administrative Control 5.5.9 has no safety significance.

Therefore, the proposed revision will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. *Would not involve a significant reduction in the margin of safety.*

A. Technical Specification Table 3.3.1-1, Item 7

As described above in Item 1, the design basis accident affected by the proposed change is the Loss of Feedwater Flow event. The Steam Generator Level-Low Reactor Trip Setpoint, in combination with the Auxiliary Feedwater Actuation System, ensures that adequate secondary side water inventory exists in both RSGs to remove decay heat following a Loss of Feedwater Flow event. To ensure that the acceptance criteria for the Loss of Feedwater Flow event are met with the RSGs, there must be at least as much mass in RSG at the Safety Analysis water level as in the OSG. The OSG Safety Analysis water level is 116.4 inches below normal water level. Using the same method to predict steam generator inventory, at this water level, OSG has 64,049 lbm water mass and RSG has 64,115 lbm water mass. Therefore, the RSG has more post-reactor trip secondary side inventory than the OSG which ensures the Loss of Feedwater event acceptance criteria are not challenged.

Therefore, the proposed revision to change the reference setpoint for steam generator low level reactor trip function does not involve a significant reduction in the margin of safety.

B. LCO 3.4.1 and Surveillance Requirement 3.4.1.3

As described above in Item 1, increasing the required minimum RCS total flow rate has no adverse impact on the safety analysis. Crediting more RCS flow in the safety analysis allows for greater flexibility in core design and operation. The increase in RCS flow associated with the RSG is within the bounds previously analyzed for the OSG. The hydraulic forces experienced around the RCS loop, including the core uplift force, are acceptable. The change is more restrictive in nature in that more RCS flow will be required to meet Surveillance Requirement 3.4.1.3 and more RCS flow ensures enhanced core heat removal. The overall core thermal margin in the safety analysis will remain essentially the same.

Therefore, the proposed revision to increase reactor coolant minimum required total flow rate does not involve a significant reduction in the margin of safety.

C. Technical Specification Administrative Control 5.5.9

As described in Item 1C above, the three sleeving options were acquired by CCNPP for economic reasons to maintain OSG thermal output by minimizing the number of tubes plugged. Therefore, deletion of these repair options from Technical Specification Administrative Control 5.5.9 has no safety significance.

Therefore, the proposed revision does not involve a significant reduction in the margin of safety.

ATTACHMENT (3)

MARKED-UP TECHNICAL SPECIFICATION PAGES

Page Nos.

3.3.1-10
3.3.1-1
3.4.1-2
5.0-18
5.0-19

Table 3.3.1-1 (page 2 of 3)
Reactor Protective System Instrumentation

FUNCTION	MODES	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Steam Generator Pressure-Low[c]	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9	≥ 685 psia
7. Steam Generator Level-Low	1, 2	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.8 SR 3.3.1.9	≥ ~~10~~ 50 inches below ~~top of feed ring~~ normal water level*
8. Axial Power Distribution-High[d]	1[e]	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9	In accordance with the COLR
9a. Thermal Margin/Low Pressure (TM/LP)[b]	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.8 SR 3.3.1.9	In accordance with the COLR

* For Unit 2, the ALLOWABLE VALUE shall remain ≥ 10 inches below top of feed ring through Cycle 14.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2200 psia;

b. RCS cold leg temperature (T_c) ≤ 548°F; and

c. RCS total flow rate ≥ ~~340,000~~ 370,000* gpm.

APPLICABILITY: MODE 1.

---------------------------- NOTE ----------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCS DNB parameter(s) not within limits.	A.1	Restore parameter(s) to within limit.	2 hours

* For Unit 2, the RCS total flow rate shall remain ≥ 340,000 gpm through Cycle 14.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure ≥ 2200 psia.	12 hours
SR 3.4.1.2	Verify RCS cold leg temperature ≤ 548°F.	12 hours
SR 3.4.1.3	Verify RCS total flow rate ≥ 370,000* ~~340,000~~ gpm.	12 hours
SR 3.4.1.4	Verify measured RCS total flow rate is within limits.	24 months

* For Unit 2, the RCS total flow rate shall remain ≥ 340,000 gpm through Cycle 14.

5.5 Programs and Manuals

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%~~
ii ~~iii.~~	ABB-Combustion Engineering leak tight sleeve wall (Unit 2 through Cycle 14 only. Not applicable for Unit 1)	28%
iii ~~iv.~~	ABB-Combustion Engineering Alloy 800 leak-limiting sleeve wall (Unit 2 through Cycle 14 only. Not applicabe for Unit 1.)	35%

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.

5.5 Programs and Manuals

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, 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.~~

    i. ~~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.

    (Unit 2 through Cycle 14 Only. Not applicable for Unit 1.)

    ii. ~~iii.~~ ABB-Combustion Engineering Alloy 800 leak-limiting sleeving as described in the Proprietary ABB Combustion Engineering Report CEN-633-P, Revision 03-P, "Steam Generator Tube Repair For Combustion Engineering Designed Plants with 3/4-.048 Inch Wall Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves, " October 1998.

    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.