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December 9, 2003

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: Increase of the Lift Setpoint of the First Bank of Main Steam Safety Valves and Increase in the Completion Time to Reset the Power Level-High Trip Setpoint

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. hereby requests an amendment to increase the maximum allowable lift setting on two main steam safety valves (MSSVs) on each unit. The proposed amendment will also increase the completion time for reducing the Power Level-High Trip setpoint.

Technical Specification Surveillance Requirement 3.7.1.1 requires that the two MSSVs with the lowest lift settings on both units be set within the range of 935 psig to 995 psig. This amendment request proposes to change the upper end of the range to 1005 psig to allow additional margin to account for setpoint drift.

Technical Specification Required Action 3.7.1.A.2 requires a reduction in the Power Level-High Trip setpoint within 12 hours if one or more required MSSVs are inoperable. This amendment request proposes to change the completion time to 36 hours, which is considered a more reasonable time to change the setpoints and/or correct an MSSV inoperability and is consistent with TSTF-235, Revision 1.

The significant hazards discussion and the technical basis for this proposed change are provided in Attachment (1). Marked up pages of the affected Technical Specification are provided in Attachment (2). Typed Technical Specification pages are provided in Attachment (3). Note that amendments approved by the Nuclear Regulatory Commission during the review period for this request may change these typed pages. The Technical Specification Bases will be changed as appropriate to support this amendment.

Our Plant Operations Safety Review Committee and Nuclear Safety Review Board have reviewed these proposed changes to the Technical Specifications and our determination of significant hazards. They have concluded that implementation of these changes will not result in an undue risk to the health and safety of the public.

We request approval of this change by November 1, 2004.

A001

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, George Vanderheyden, being duly sworn, state that I am Vice President - 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 St. Mary's, this 9th day of December, 2003.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

March 25, 2007
Date

GV/EMT/bjd

Attachments: (1) Technical Basis and Significant Hazards Consideration
(2) Marked up Technical Specification Pages
(3) Final Technical Specification Pages

cc: J. Petro, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
G. S. Vissing, NRC
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Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

TECHNICAL BASIS AND

SIGNIFICANT HAZARDS CONSIDERATION

TABLE OF CONTENTS

ATTACHMENT (1)

TECHNICAL BASIS AND SIGNIFICANT HAZARDS CONSIDERATION

1.0 DESCRIPTION

This letter is a request to amend Operating License Numbers DPR-53 and DPR-69 for Calvert Cliffs Units 1 and 2. The proposed change will revise the operating licenses, allowing additional margin to account for setpoint drift by changing the upper end of the setpoint of the first main steam safety valves (MSSVs) to lift to 1005 psig. The proposed change will also change the completion time to reset the Power Level-High Trip setpoint to 36 hours, which is considered a more reasonable time for correction of an MSSV inoperability.

2.0 PROPOSED CHANGE

Technical Specification Surveillance Requirement 3.7.1.1 requires that the two MSSVs with the lowest lift settings on both units be set within the range of 935 psig to 995 psig. This amendment request proposes to change the upper-end of the range to 1005 psig to allow additional margin to account for setpoint drift.

Technical Specification Required Action 3.7.1.A.2 requires a reduction in the Power Level-High Trip setpoint within 12 hours if one or more required MSSVs are inoperable. This amendment request proposes to change the completion time to 36 hours, which is considered a more reasonable time to change the setpoints and/or correct an MSSV inoperability and is consistent with TSTF-235, Revision 1.

3.0 BACKGROUND

Main Steam Safety Valves

The primary purpose of the MSSVs is to provide overpressure protection for the main steam system (110% of 1015 psia) in both Calvert Cliffs units. Eight MSSVs are located on each main steam header; there are two main steam headers in each unit. The valves also provide protection against over pressurizing the reactor coolant pressure boundary (110% of 2500 psia) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink - the main condensers and the Circulating Water System - is not available.

The functional design of the MSSVs includes staggered setpoints, as illustrated by Technical Specification Table 3.7.1-2. Staggered setpoints reduce the potential for valve chattering when there is insufficient steam pressure to fully open all valves following a turbine trip. They also minimize the number of valves actuated during those transients where less than maximum relief capacity is required. The setpoint testing for MSSVs complies with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Article IWV-3510.

The MSSV rated capacity passes full steam flow at 102% rated thermal power (100% plus 2% for instrument error) with all eight valves fully open. The capacity design basis for the valves comes from the ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 components. The purpose of the valves is to limit secondary system pressure to $\leq$ 110% of system design pressure when passing 100% of design steam flow.

Power Level - High Channels

The Power Level-High Trip uses Q power (the higher of NI power and delta-T power) as its only input. The purpose of the Power Level-High Trip is to trip the reactor if power exceeds the set value. Therefore, the trip setpoint must be changed as reactor power is reduced due to a reduction in the number of operable MSSVs.

An alternative to restoring inoperable MSSVs to operable status is to reduce reactor power so that the operable MSSV relieving capacity meets Code requirements for the power level. Technical Specification 3.7.1, Required Action A.1 requires plant operators to reduce power to a level provided in Technical Specification Table 3.7.1-1 when one or more MSSVs are inoperable. This table specifies a power level dependant on the number of MSSVs remaining operable.

The number of inoperable MSSVs will also determine the necessary reactor trip settings of the Power Level-High Trip channels. Technical Specification 3.7.1, Required Action A.2 directs the reduction of the Power Level-High Trip channels to the setpoints provided in Technical Specification Table 3.7.1-1.

4.0 TECHNICAL ANALYSIS

Proposed Change to MSSV Pressure Setting

Updated Final Safety Analysis Report (UFSAR) Chapter 14 safety analyses were reviewed as part of the proposed change to the maximum allowable lift settings of the two MSSVs to lift.

Safety analyses for the UFSAR Chapter 14 excess load events and large break loss-of-coolant accidents (LOCAs) were reviewed and found unaffected by the proposed change. In these events, the secondary system undergoes decreasing pressure during the transient. This pressure change occurs so quickly that either the secondary system does not play a role in determining the consequences, or the events are not sensitive to the proposed changes to the MSSV lift setpoints.

The Chapter 14 events that are affected by the proposed change are: 14.2 – CEA Withdrawal; 14.5 - Loss of Load Event; 14.6 – Loss-of-Feedwater Flow Event; 14.10 – Loss-of-Non-Emergency AC Power; 14.12 - Asymmetric Steam Generator Event; 14.17.3 – Base Small Break LOCA Analysis; and 14.26 - Feedline Break Event). These events required reanalysis or reevaluation. The events, as described below, continue to meet applicable specified acceptable fuel design limits (SAFDLs) criteria for departure from nucleate boiling and linear heat rate, and calculated primary and secondary pressures do not exceed the limit of 110% of their respective design pressures.

Control Element Assembly Withdrawal

The control element assemblies (CEAs), in a pre-programmed sequence, are used to control (dampen) xenon oscillations and rapidly control core power. The CEAs also provide the required negative reactivity for shutdown during design basis events.

A CEA Withdrawal Event (UFSAR Section 14.2) is defined as any event caused by a single malfunction in the reactor regulating system or control element drive mechanism control system that results in a continuous sequential CEA group withdrawal. This event can approach the SAFDLs and the RCS pressure upset limit of the departure from nucleate boiling ratio and linear heat generation rate. Initial margins maintained by Technical Specifications in conjunction with the reactor protective system (Variable High Power Trip or Axial Flux Trip) ensure that these design limits will not be exceeded. Since no pin failures are postulated to occur, the site boundary dose criteria in 10 CFR Part 100 guidelines will not be approached.

The event was reanalyzed to ensure that calculated primary and secondary pressures do not exceed the limit of 110% of design pressure. The event results in an addition of reactivity, an increase in core power, and an increase in primary and secondary pressures. Although the higher allowable lift setting for the first two MSSVs could delay the opening of the valves, the calculated primary and secondary pressures

do not exceed the limit of 110% of design pressure. Additionally, the radiological consequences associated with the event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Loss of Load

A Loss of Load Event (UFSAR Section 14.5) is any event that results in a reduction in the steam generator heat removal capacity through the loss of secondary steam flow. Closure of all main steam isolation valves, turbine stop valves, or turbine control valves will cause a Loss of Load Event. Of the three types of valves in the steam lines between the steam generator and the high pressure turbine, the turbine stop valves have the quickest closure time.

The most limiting Loss of Load Event is a turbine trip without a concurrent reactor trip or an inadvertent closure of the turbine stop valves at hot full power. A turbine trip would result in the closure of the turbine stop valves.

In the case of inoperable MSSVs, the loss of load safety analysis credits reactor power reduction. The Power Level-High Trip is not credited in the loss of load analysis but reducing this trip setpoint within the Technical Specification Completion Time ensures the thermal power limit supported by the safety analysis is met.

The Loss of Load Event was reanalyzed to ensure that calculated primary and secondary pressures do not exceed the limit of 110% of design pressure. The Loss of Load Event is a decreased heat removal event. Although the higher allowable lift setting for the first two MSSVs could delay the opening of the valves, the calculated primary and secondary pressures do not exceed 110% of design pressure. Additionally, the radiological consequences associated with the Loss of Load Event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Loss-of-Feedwater Flow

The primary function of the feedwater regulating system is to maintain the liquid inventory in the steam generator at the normal operating level. The feedwater regulating valve controller, which is part of a three-element control system, automatically adjusts the position of the regulating valves to modulate the feedwater flow. During normal operation, the controller matches the feedwater flow to the steam flow.

The main feedwater trains consist of three electric motor-driven condensate pumps in parallel that take suction on the three condensers and discharge to the three parallel electric motor-driven condensate booster pumps. These pumps deliver the condensate to two parallel, steam turbine-driven steam generator feed pumps.

Check valves in the condensate and feedwater system reduce the likelihood of a total loss-of-feedwater and blowdown of both steam generators due to a pipe break in the condensate or feedwater systems.

A Loss-of-Feedwater Flow Event (UFSAR Section 14.6) is defined as a reduction in feedwater flow to a steam generator without a corresponding reduction in steam flow from the steam generator. The closure of the feedwater regulating valves, the loss-of-condensate or feedwater pumps, or a pipe break in the condensate or feedwater system during steady-state operation would result in this event.

The most limiting Loss-of-Feedwater Flow Event at hot full power is an inadvertent closure of both feedwater regulating valves. An instantaneous closure of the regulating valves would cause the largest steam and feedwater flow mismatch and result in the most rapid reduction in the steam generator inventory.

An analysis of the Loss-of-Feedwater Event considering the installation of the new replacement steam generators was previously submitted to the Nuclear Regulatory Commission (NRC) in Reference (1) and approved by Reference (2). This analysis conservatively used an increased initial allowable lift setting of 1020 psia (1005 psig) for the first two MSSVs and considered increased pressure in the RCS and steam generators and steam generator inventory depletion. The analysis concluded that the calculated primary and secondary pressures do not exceed 110% of design pressure. Additionally, the radiological consequences associated with the Loss-of-Feedwater Event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Loss-of-Non-Emergency AC Power Event

Loss-of-Non-Emergency AC Power (LOAC) Event is defined as a loss of the plant's 500 kV/13 kV service transformers. The most limiting LOAC event is a loss of turbine load at hot full power with offsite AC power unavailable.

The immediate system response is very similar to a loss of load, loss of coolant flow, or loss-of-feedwater flow. A reactor trip occurs early (about one second) in the transient. Due to the unavailability of the condenser, the steam dump and turbine bypass system is inoperable. The RCS and main steam system pressurize as the steam generators are unable to remove all of the core heat due to the reduction in RCS flow. The MSSVs lift to limit the pressure increase and remove decay heat.

The LOAC event was re-evaluated to ensure that the calculated primary and secondary pressures do not exceed the limit of 110% of design pressure. Although the higher allowable lift setting for the first two MSSVs could delay the opening of the valves, the calculated primary and secondary pressures do not exceed 110% of design. Additionally, the radiological consequences associated with this event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Asymmetric Steam Generator Event

A primary function of the steam generators is to remove heat from the RCS. Any perturbation within the steam generators will affect the RCS due to the close coupling of both systems.

An Asymmetric Steam Generator Event (UFSAR Section 14.12) is defined as any initiator that affects only one of the two steam generators. A loss of load, an excess load, a loss-of-feedwater, or excess feedwater to only one steam generator would result in an Asymmetric Steam Generator Event.

Asymmetric Steam Generator Events, which are the result of a malfunction of one steam generator, cause a non-uniform reactor core inlet temperature distribution. The non-uniform core inlet temperature distribution in conjunction with the moderator temperature reactivity feedback produces asymmetric local power peaking in the core.

The most limiting Asymmetric Steam Generator Event is a loss of load to one steam generator. An asymmetric loss of load would produce the largest core inlet temperature differential across the core. Based on a negative moderator temperature coefficient, the RCS temperature tilt across the core will cause an increase in local core power peaking and an approach to the SAFDLs. Action by the Reactor Protective System will prevent exceeding the SAFDLs.

The Asymmetric Steam Generator Event was reanalyzed to ensure that calculated primary and secondary pressures do not exceed the limit of 110% of design pressure. Although the higher allowable lift setting for the first two MSSVs could delay the opening of the valves, the calculated primary and secondary

pressures do not exceed 110% of design pressure. Additionally, the radiological consequences associated with this event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Small Break Loss-of-Coolant Accident

The results of a small break LOCA, with an increase to the upper end of the setpoint range of 10 psi (995 psig to 1005 psig) for the first two MSSVs, were previously submitted to the NRC in Reference (3).

The small break LOCA analysis submitted in Reference (3) uses Supplement 2 to CENPD-137 Evaluation Model, which was accepted by the NRC for licensing applications in Combustion Engineering-designed pressurized water reactors, such as Calvert Cliffs. Westinghouse and Calvert Cliffs have on-going processes that ensure that the as-operated plant values for peak cladding temperature-sensitive parameters remain bounded by the values used in the analysis detailed in Reference (3).

Feedline Break Event

The Feedline Break Event (UFSAR Section 14.26) is initiated by a break in the Main Feedwater System piping and may occur as a result of thermal stress or cracking in the main feed line. The guillotine-type break assumed in the Safety Analysis is the most adverse transient scenario and its probability of occurrence is extremely low.

The feedwater line break is a decreased heat removal event that results in increasing primary and secondary pressures. The event was reanalyzed with and without loss-of-AC power on turbine trip to ensure that calculated primary and secondary pressures do not exceed the limit of 110% of design pressure. Although the higher allowable lift setting for the first two MSSVs could delay the opening of the valves, the calculated primary and secondary pressures do not exceed 110% of design pressure. Additionally, the radiological consequences associated with the Feedline Break Event continue to be negligible when compared to the 10 CFR Part 100 guidelines.

Proposed Change to Power Level-High Trip Completion Time

If one or more MSSVs are declared inoperable, reactor power must be reduced to a level mandated by Table 3.7.1-1 (Required Action 3.7.1.A.1). Four hours is allowed to complete this action. The 12-hour completion time for Required Action 3.7.1.A.2 allows an additional eight hours to reduce the Power Level-High Trip setpoint after power has been reduced. This submittal proposes to change the completion time for Required Action 3.7.1.A.2 to 36 hours to better allow for preparation and adjustment of each channel. This change will allow 32 hours in addition to the four-hour completion time of Required Action 3.7.1.A.1 and is consistent with TSTF-235, Revision 1, which was approved by the NRC January 11, 1999.

As stated in TSTF-235, Revision 1, a completion time of 36 hours for Required Action 3.7.1.A.2 is based on a reasonable time to correct the MSSV inoperability, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

Probabilistic Risk Analysis

Increasing the lift setpoint of the first two MSSVs and increasing the Power Level-High Trip setpoint completion time have been reviewed for their effect on plant core damage frequency and large early release probabilities. Our analyses shows that these proposed changes have a negligible impact and are well within the criteria of Regulatory Guides 1.174 and 1.177.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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.*

   This license amendment request proposes to increase the upper range of the relief setting of the first two Main Steam Safety Valves (MSSVs) by 10 psi. The MSSVs are not accident initiators. They are credited with relieving secondary system pressure and act as a heat sink for the Reactor Coolant System (RCS) when the preferred heat sink is not available. Increasing the upper end of the setpoint for the first two MSSVs to lift does not affect the steam relieving capacity of the total or any combination of MSSVs that lift. This proposed amendment does not install any new components or change the physical characteristics of the MSSVs. Therefore, the change does not involve a significant increase in the probability of an evaluated accident.

   The Updated Final Safety Analysis Report Chapter 14 safety analyses were reviewed considering the change to the upper end of the lift settings range of the first two MSSVs. The analyses show that increasing the upper end of the lift setting range does not exceed the pressure limits of the reactor coolant or main steam systems, nor the radiological consequences anticipated by the safety analyses. Therefore, the change will not involve a significant increase in the consequences of an evaluated accident.

   This proposed amendment will also increase the Technical Specification Completion Time to reset the Power Level-High Trip from 12 hours to 36 hours. The purpose of the Power Level-High Trip is to trip the reactor if reactor power exceeds a set value, and is required by Technical Specifications to be reset according to the number of MSSVs remaining operable. The trip is not an accident initiator but is a signal that responds to an accident condition. Therefore, the change does not involve a significant increase in the probability of an evaluated accident.

   Reducing the setpoint of the Power Level-High Trip within the time allotted by Technical Specifications provides additional assurance that the MSSVs will be able to perform their design function by keeping the reactor power within the ability of the MSSVs to relieve steam volume. There is a low probability of a transient that could result in steam generator overpressure during the proposed 36 hours to reset the Power Level-High Trip. Therefore, this change does not involve a significant increase in the consequences of an evaluated accident.

   Therefore, this proposed license amendment does not significantly increase 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.*

   The proposed amendment will increase the upper end of the lift pressure for the first two MSSVs and increase the Technical Specification Completion Time to reset the Power Level-High Trip setpoint.

The proposed amendment does not involve a physical alteration of the plant or change the plant configuration. It does not require any new or unusual operator actions. The amendment does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated. It does not introduce any new failure modes.

Therefore, this proposed license amendment does 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.*

The margin of safety in this case is that the MSSVs release sufficient steam to relieve pressure in the secondary system and to act as a heat sink to prevent over-pressurization of the RCS when the preferred heat sink is not available. Increasing the upper end of the setpoint for the first two MSSVs to lift does not affect the steam relieving capacity of the total or any combination of MSSVs that lift. Potential delay in the opening of the first two MSSVs does not result in exceeding the pressure limits of the reactor coolant or main steam systems.

Reducing the Power Level-High Trip setpoint within the specified time limit provides additional assurance that the MSSVs will be able to perform their design function by keeping the reactor power within the ability of the MSSVs to relieve steam volume. A completion time of 36 hours to lower the Power Level-High Trip setpoint is based on a reasonable time to correct the MSSV inoperability, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

Therefore, this proposed license amendment does not involve a significant reduction in the margin of safety.

6.0 ENVIRONMENTAL CONSIDERATION

We have determined that operation with the proposed amendment would not result in any significant change in the types or amounts of any effluents that may be released offsite, nor would it result in any significant increase 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 proposed amendment.

7.0 PRECEDENT

Millstone Nuclear Power Station Approved May, 2003

Millstone proposed a 36-hour completion time for resetting the Power Level-High Trip setpoint. The proposal was based on Technical Specification Task Force-235 and that the proposed completion time is a reasonable time to correct the MSSV inoperability, the time required to perform power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. The NRC approved that proposal.

8.0 REFERENCES

1. Letter from Mr. C. H. Cruse (CCNPP) to Document Control Desk (NRC), dated November 19, 2001, "License Amendment Request: Reanalysis of the Loss of Feedwater Event"

2. Letter from Ms. Donna Skay (NRC) to Mr. C. H. Cruse (CCNPP), dated February 26, 2002, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Reanalysis of Loss of Feedwater Event (TAC Nos. MB3442 And MB3443)"

3. Letter from Mr. C. H. Cruse (CCNPP) to Document Control Desk (NRC), dated May 9, 2002, "10 CFR 50.46 30-Day Report for Changes to the Calvert Cliffs Nuclear Power Plant Emergency Core Cooling System Performance Analysis"

ATTACHMENT (2)

MARKED UP TECHNICAL SPECIFICATION PAGES

3.7.1-1

3.7.1-4

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1. AND A.2 Reduce the Power Level-High Trip setpoint in accordance with Table 3.7.1-1.	4 hours ~~12~~ 36 hours

Table 3.7.1-2
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING[1] (psig)
Steam Generator #1	Steam Generator #2	
RV-3992	RV-4000	935-995 1005
RV-3993	RV-4001	935-995 1005
RV-3994	RV-4002	935-1035
RV-3995	RV-4003	935-1035
RV-3996	RV-4004	935-1050
RV-3997	RV-4005	935-1050
RV-3998	RV-4006	935-1050
RV-3999	RV-4007	935-1050

(1) Lift settings for a given steam line are also acceptable if any two valves lift between 935 and 995 1005 psig, any two other valves lift between 935 and 1035 psig, and the four remaining valves lift between 935 and 1050 psig.

ATTACHMENT (3)

FINAL TECHNICAL SPECIFICATION PAGES

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours
	AND	
	A.2 Reduce the Power Level-High Trip setpoint in accordance with Table 3.7.1-1.	36 hours

Table 3.7.1-2
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING[1] (psig)
Steam Generator #1	Steam Generator #2	
RV-3992	RV-4000	935-1005
RV-3993	RV-4001	935-1005
RV-3994	RV-4002	935-1035
RV-3995	RV-4003	935-1035
RV-3996	RV-4004	935-1050
RV-3997	RV-4005	935-1050
RV-3998	RV-4006	935-1050
RV-3999	RV-4007	935-1050

(1) Lift settings for a given steam line are also acceptable if any two valves lift between 935 and 1005 psig, any two other valves lift between 935 and 1035 psig, and the four remaining valves lift between 935 and 1050 psig.