NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

April 25, 2006

L-MT-06-034
10 CFR 50.90

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

Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

References:
1. Letter from Nuclear Management Company, LLC, to Document Control Desk, "License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS)," dated June 29, 2005
2. Letter from Nuclear Management Company, LLC, to Document Control Desk, "Supplement to License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS)," dated April 25, 2006

Copy of Applicable Portions of the U.S. Nuclear Regulatory Commission and Monticello Nuclear Generating Plant Improved Technical Specifications Conversion Website

By References 1 and 2, Nuclear Management Company, LLC (NMC) submitted an application and supplement to amend the Technical Specifications of Monticello Nuclear Generating Plant (MNGP), Facility Operating License DPR-22, revising the current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications (ISTS) as described in NUREG-1433, "Standard Technical Specifications General Electric Plants BWR/4," Revision 3, and certain generic changes to the NUREG.

The purpose of this letter is to provide a copy of applicable portions of the U.S. Nuclear Regulatory Commission (NRC) and MNGP ITS Conversion Website (Enclosure) suitable for posting on the MNGP docket, Docket No. 50-263. This information was provided by NMC on the NRC and MNGP ITS Conversion Website. This information was used by NMC in development of Reference 2, and documents the NRC review process for approving the requested amendments to the MNGP Facility Operating License. Tracking-type questions or editorial-type questions from the joint NRC and

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MNGP ITS Conversion Website are not included in this letter. The Enclosure is arranged chronologically by the dates of the original NRC questions, and includes the applicable NRC questions, NMC response, and any attached electronic documentation, with the exception of the draft ITS submittal markup pages. These pages are not included since the changes have subsequently been provided in Reference 2.

As part of the NRC review of Reference 1, NRC questions were provided using the NRC and MNGP ITS Conversion Website. The NRC and MNGP ITS Conversion Website was developed specifically to expedite NRC review and minimize the time delay between review and posting of NRC questions, development and posting of NMC responses, and acceptance and closure of each identified NRC question by the responsible NRC reviewer. As agreed to between the NRC and NMC, entry of NRC questions and NMC responses to the NRC and MNGP ITS Conversion Website was protected so that only the NRC reviewers and NMC staff can enter information into the associated database fields for each item. In addition, only the NRC reviewers and NMC staff can attach additional electronic documentation associated with an NRC question or NMC response. However, the public could fully access all information on the NRC and MNGP ITS Conversion Website at any time during the NRC review process up until issuance of the NRC Safety Evaluation, including NRC questions, NMC response, and any attached electronic documentation.

This letter makes no new commitments or changes to any existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on April 25, 2006

John T. Conway

J. T. Conway
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, ~~LLC~~

Enclosure: Applicable Portions of the U.S. Nuclear Regulatory Commission (NRC) and Monticello Nuclear Generating Plant (MNGP) Improved Technical Specification (ITS) Conversion Website cc (w/o enclosure):

Administrator, Region III, USNRC
ITS Project Manager, Monticello, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

Enclosure

Applicable Portions of the U.S. Nuclear Regulatory Commission (NRC) and
Monticello Nuclear Generating Plant (MNGP)
Improved Technical Specification (ITS) Conversion Website

Close

NRC ITS TRACKING

NRC Reviewer

ID	200509160934 Conference Call Requested? No
Category	Major Technical
ITS Information	ITS Section: 3.7 ITS Number: 3.7.2 DOC Number: M.1 JFD Number: 7 Bases JFD Number: 1 Page Number(s): 29
Comment	Completion time for Action A: 3.7.2 is a new TS section and the completion time in the STS is 72 hours. You are proposing to extend the completion time from 72 hours to 7 days. Since this proposed TS change is different from both the CTS and the STS it should be a beyond scope item (BSI). Provide this BSI for the Techniical Branch review.
Issue Date	09/16/2005

Close Date	02/24/2006 Resolution requires change to: None Other Docket Response Required? No

Responses

NRC Response by Pete Hearn on 02/02/2006	In your response you state that the ESW system provides cooling water to the core sprays (CS) and the residual heat removal systems (RHR). Section 3.5.1 Action Statement B allows 7 days outage for 1 CS or 1 RHR train. If the allowed outage for the ESW train is 7 days, it appears that Action Statement B of 3.5.1 would be violated since both the CS and RHR train would be inoperable for 7 days. Provide a justification for extending ESW outage time from 72 hours to 7 days.
Licensee Response by Jerry Jones on 01/20/2006	During the weekly NRC/Monticello phone conference where updates on the ITS conversion are provided, the NRC stated that this issue has not been sent to the Technical Branch for review. However, the NRC requested additional information as to why the Improved Standard Technical Specifications (ISTS) Completion Time of 72 hours was not being adopted. The Monticello Current Technical Specifications (CTS) do not provide any requirements for the Emergency Service Water (ESW) System and Ultimate Heat Sink (UHS). Therefore, the addition of ITS 3.7.2 is a more restrictive change. The Completion Time of ISTS 3.7.2 Required Action E.1 (Attachment 1, Volume 12, Rev. 0, Page 29 of 161) has been changed from 72 hours to 7 days, as shown in Improved Technical Specifications (ITS) 3.7.2 Required Action A.1 Completion Time. The ISTS Bases for ACTION E.1, second paragraph (Page 37 of 161) states, in part, that the 72 hour Completion Time is based on being consistent with the allowed Completion Time for restoring an inoperable diesel generator. At Monticello, the ESW System does not provide cooling to the emergency diesel generators

	(EDGs); they receive cooling from the EDG-ESW System. The EDG-ESW System requirements are specified in ITS 3.7.3 (Page 49 of 161). As stated in the Background section of the ITS 3.7.2 Bases (Page 34 of 161), the ESW System provides cooling water to the core spray (CS) and residual heat removal (RHR) pump coolers. Therefore, the Completion Time was changed to be consistent with the time to restore an inoperable RHR or CS pump (as stated in Justification for Deviation (JFD) 7 (Page 32 of 161). Therefore, Monticello believes that this proposed Completion Time is acceptable and no changes to the ITS submittal are necessary.
Licensee Response by Jerry Jones on 02/21/2006	Based on the NRC reviewer's response to the Monticello response, and a further phone conversation with the NRC reviewer discussing the NRC reviewer's concerns, Monticello will revise the Improved Technical Specifications (ITS) submittal to adopt the 72 hour Completion Time for ITS 3.7.2 ACTION A (Attachment 1, Volume 12, Rev. 0, Page 29 of 161). The proposed change is shown in the attachment to this response.

Date Created: 09/16/2005 09:34 AM by Pete Hearn
Last Modified: 02/24/2006 09:49 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200509161013** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	ITS Section: **3.7** ITS Number: **3.7.2** DOC Number: **A.1** JFD Number: **4** Bases JFD Number: **1** Page Number(s): **28**
Comment	**Ultimate Heat Sink Maximum Allowed Temperature: You propose to delete the UHS maximum allowed Temp from the TS LCO. It appears that the UHS T satisfies the requirements of Criterian 2 of 10 CFR 50.36. Provide a justification for deleting the UHS T from the LCO that addresses the requirements of Criterean 2.**
Issue Date	**09/16/2005**

Close Date	**02/13/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

NRC Response by Pete Hearn on 01/19/2006	Page 28 of 161 and Page 30 of 161 You are proposing to eliminate STS Actions D and F form the BV TS. This removes the definition for UHS Temperature operability and the requirements to shutdown the plant when the UHS Temperature Requirements are not met. This propsal is in contrast to TSTF-330. Justify not conforming to TSTF-330.
Licensee Response by Jerry Jones on 10/11/2005	Improved Standard Technical Specifications (ISTS) 3.7.2 ACTION D (Attachment 1, Volume 12, Rev. 0, Page 28 of 161) requires entry when water temperature of the Ultimate Heat Sink (UHS) is greater than the bracketed limit of 90 degrees F but less than or equal to a bracketed upper limit (no limit provided in the ISTS) and requires the verification that the water temperature of the UHS is less than or equal to 90 degrees F averaged over the previous 24 hour period. This verification is required once per hour. A Reviewer's Note is included for this ACTION (in the Condition) that states that the bracketed upper temperature limit is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit. ISTS 3.7.2 ACTION D was added to the ISTS in accordance with Technical Specification Task Force (TSTF) 330-A, Rev. 3. The TSTF stated in the first paragraph of the Justification section that "The existing UHS requirements introduce the possibility of additional plant shutdown transients. Potential plant shutdown transients could be reduced by the additional Required Action to average UHS water temperature on a more frequent basis. A plant shutdown would be required if the averaged UHS water temperature limit

were exceeded or if the maximum temperature limit were exceeded. With the water temperature of the UHS exceeding the SR limit but less than a maximum allowed value (specified in the Required Action), the design basis assumptions associated with initial UHS temperatures are bounded provided the temperature of the UHS averaged over the previous 24 hour period is less than the SR limit. With the water temperature of the UHS greater than the SR limit, long term cooling capability to dissipate the heat of an accident safely may be affected. Therefore, to ensure long term cooling capability when UHS water temperature is above the SR limit, more frequent monitoring and averaging of the temperature over the previous 24 hour period is required." Thus, the TSTF was providing an allowance to exceed the UHS temperature limit specified in ISTS SR 3.7.2.3 (Page 30 of 161), provided the 24-hour averaged value did not exceed the SR limit and a higher maximum limit was not exceeded. Furthermore, in order to adopt the TSTF allowance of ACTION D, the TSTF (in the remaining paragraphs of the Justification section) required that the Licensees who wish to adopt this change to confirm that the following conditions, which form the basis for acceptance of the UHS temperature averaging approach, are satisfied. a. The UHS is not relied upon for immediate heat removal (such as to prevent containment overpressurization), but is relied upon for longer-term cooling such that the temperature averaging approach continues to satisfy the accident analysis assumptions for heat removal over time. b. When the UHS is at the proposed maximum allowed value of [ ] F, equipment that is relied upon for accident mitigation, anticipated operational occurrences, or for safe shutdown, will not be adversely affected and are not placed in alarm condition or limited in any way at this higher temperature. c. Plant-specific assumptions, such as those that were credited in addressing station blackout and Generic Letter 96-06, have been adjusted (as necessary) to be consistent with the maximum allowed UHS temperature of [ ] F that is proposed. d. Cooling water that is being discharged from the plant (either during normal plant operation, or during accident conditions), does not affect the UHS intake water temperature (typical of an infinite heat sink, but location of the intake and discharge connections, and characteristics of the UHS can have an impact). In addition, the license amendment request must include a discussion of these conditions, and confirm that the conditions are satisfied. Any exceptions must be identified and justified in the amendment request, and factored into the plant-specific UHS limitations that are proposed. NMC has not performed the above actions since it does not desire to adopt the allowance of ISTS 3.7.2 ACTION D within the scope of the ITS submittal. Therefore, ISTS 3.7.2 ACTION D has not been included in the Monticello Improved Technical Specifications (ITS). The justification for not including the ACTION is presented in Justification for Deviations (JFD) 4 (Page 32 of 161), which states "The bracketed ISTS ACTION D has been deleted as it is not part of the plant specific ITS. The 90 degree F limit in ITS SR 3.7.2.2 is the maximum water temperature assumed in the accident analysis. Therefore, when 90 degrees F is exceeded, the UHS is inoperable and ISTS 3.7.2 ACTION F (ITS 3.7.2 ACTION B) must be taken." Specifically, ITS LCO 3.7.2 requires two Emergency Service Water (ESW) subsystems and the UHS to be OPERABLE (Page 28 of 161). ITS SR 3.7.2.2 (Page 30 of 161) includes the requirement to verify the average water temperature (averaged based on location of temperature samples, not over a time period) of UHS is less than or equal to 90 degrees F. The ITS 3.7.2 Bases, LCO section (Page 35 of 161), states that the OPERABILITY of the UHS is based on having a maximum water temperature of 90 degrees F. ITS 3.7.2 ACTION B (Page 30 of 161) requires entry when the UHS is inoperable (e.g., UHS temperature not within the limit of ITS SR 3.7.2.2) and requires the plant to be in MODE 3 within 12 hours and MODE 4 in 36 hours. The average

	water temperature of 90 degrees F is the temperature required to support the Monticello safety related components. Therefore, the Monticello analysis will be met as long as the UHS temperature is less than or equal to 90 degrees F, and the bracketed allowance of ISTS 3.7.2 ACTION D is not desired to be included in the Monticello ITS.
Licensee Response by Jerry Jones on 02/10/2006	To clarify our previous response to this question, Improved Standard Technical Specification (ISTS) 3.7.2 ACTION D (Attachment 1, Volume 12, Rev. 0, Page 28 of 161) was added to the ISTS (in Revision 2 of NUREG-1433) in accordance with Technical Specification Task Force (TSTF) 330-A, Rev. 3. The TSTF provides an allowance to exceed the UHS temperature limit specified in ISTS SR 3.7.2.3 (Page 30 of 161), provided the 24-hour averaged value did not exceed the SR limit and a higher maximum limit was not exceeded. NMC has not adopted the allowance provided in TSTF-330 (ISTS 3.7.2 ACTION D) in the Monticello Improved Technical Specifications (ITS) submittal. With respect to the UHS temperature requirements, the Monticello ITS submittal reflects the ISTS as written prior to Revision 2 (i.e., Revision 1).

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NRC ITS TRACKING

NRC Reviewer

ID	200509191552 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.7 ITS Number: 3.7.4 DOC Number: A.1 JFD Number: 3 Bases JFD Number: 1 Page Number(s): 78
Comment	In SR 3.7.4.4 you propose to use the term positive pressure and not state in the TS the actual pressure provided by the CREF. You relied on a 1989 SER (Justification 3 ON PAGE 79) for justification but the SER approves the positive pressure values that the CREF provides but does not approve leaving those values out of the TS. 10 CFR 50.36 requires that the actual positive pressure be stated in the TS. Provide the preesures used in the dose analysis in the 1989 SER in SR 3.7.4.4.
Issue Date	09/19/2005

Close Date	01/18/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/11/2005	Improved Standard Technical Specifications (ISTS) SR 3.7.4.4 (Attachment 1, Volume 12, Rev. 0, Page 78 of 161) states "[Verify each [MCREC] subsystem can maintain a positive pressure of greater than or equal to [0.1] inches water gauge relative to the [turbine building] during the [pressurization] mode of operation at a flow rate of less than or equal to [400] cfm." Monticello has not incorporated the value of positive pressure in Improved Technical Specifications (ITS) SR 3.7.4.4 as justified in Justification for Deviation (JFD) 3 (Page 79 of 161). JFD 3 states "ISTS SR 3.7.4.4 specifies a bracketed positive pressure criterion of 0.1 inches water gauge relative to the turbine building. ITS SR 3.7.4.4 only requires maintaining a positive pressure relative to adjacent areas. This difference was accepted by the NRC in a letter dated May 30, 1989 from John F. Stepano (NRC) to Mr. Musolf (NSPC) and discussed in Section 2, page 7, of the associated Safety Evaluation." In Section 2, page 7, part j, of the referenced NRC Safety Evaluation, the NRC made the following observation of the test: "The staff observed the positive pressure measurements of the control room envelope during an emergency filter train operation test at the Monticello plant. There were four pressure gauges measuring pressure differentials for five locations of the control room envelope. The results ranged from 0.12 to 0.005-inch water positive pressure relative to the adjacent area of the control room envelope with the lowest reading occurring at the TSC. The licensee explained that the TSC has an outside wall with a row of windows having sealed glass panels, while other parts of the

	control room envelope are adjacent to building interiors. The licensee found that it is unworkable to maintain a 0.125-inch water positive pressure in the TSC for all expected wind velocities. The licensee has performed a preliminary calculation based on a 25 mph wind. Under this condition, there will be 2-3 cfm unfiltered inleakage from the windows, which will increase the control room envelope dose by less than 0.2 rem. This increase is not significant compared with the calculated dose. The staff concludes that the licensee's proposed TS for ensuring positive pressure in the control room is acceptable since the Monticello control room envelope is capable of preventing any significant unfiltered inleakage." The proposed Monticello Technical Specifications included in the Northern States Power Company (NSPC) amendment request included CTS 4.17.B.2.b.(3), which stated "Verify on a simulated high radiation signal, the train switches to the pressurization mode of operation and the control room is maintained at a positive pressure with respect to adjacent areas at the design flow rate of 1000 cfm (plus or minus 10%)." The NRC issued this new requirement in Amendment 65, of which the above referenced Safety Evaluation discussion is a part. Therefore, Nuclear Management Company, LLC (NMC) believes this is an endorsement of Current Technical Specifications (CTS) 4.17.B.2.c.(3) (Page 67 of 161), which does not include a value for positive pressure. Note, the CTS reference for this requirement was changed from 4.17.B.2.b.(3) to 4.17.B.2.c.(3) in a later amendment. This allowance was maintained in ITS SR 3.7.4.4. In addition, NMC reviewed 10 CFR 50.36(c)(2) and (3) and did not find anything in these regulations requiring the actual positive pressure to be stated in the Technical Specifications.

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NRC ITS TRACKING

NRC Reviewer

ID	200509200942 Conference Call Requested? No
Category	**Major Technical**
ITS Information	**ITS Section:** 3.7 **ITS Number:** 3.7.7 **DOC Number:** M.1 **JFD Number:** 3 **Bases JFD Number:** 1 **Page Number(s):** 137
Comment	**SR 3.7.7.1: The frequency for the verification in the STS is 31 days and the CTS does not address this frequency. You are proposing a frequency of 92 days in the ITS which is different than the frequency in the CTS and the STS. This proposal meets the definition of a BSI; therefore, submit this proposal as a BSI in order for the Technical NRR staff to review and evaluate your proposal.**
Issue Date	**09/20/2005**

Close Date	**02/02/2006** Resolution requires change to: JFD Other Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 01/20/2006	During the weekly NRC/Monticello phone conference where updates on the ITS conversion are provided, the NRC stated that this issue has not been sent to the Technical Branch for review. However, the NRC requested additional information as to why the Improved Standard Technical Specifications (ISTS) Surveillance Frequency of 31 days was not being adopted. The Monticello Current Technical Specifications (CTS) do not provide any requirements for the turbine bypass valves. Therefore, the addition of ITS 3.7.7, including SR 3.7.7.1, is a more restrictive change. The Frequency of ISTS SR 3.7.7.1 (Attachment 1, Volume 12, Rev. 0, Page 137 of 161) has been changed from 31 days to 92 days, as shown in Improved Technical Specifications (ITS) SR 3.7.7.1. Currently, this Surveillance, cycling each main turbine bypass valve, is being performed in a plant procedure on a quarterly basis. Therefore, Monticello changed the ISTS SR 3.7.7.1 Frequency to be consistent with the current Frequency specified in the plant procedure. This proposed Frequency has been shown to be acceptable, as stated in Justification for Deviation (JFD) 3 (Page 139 of 161). Therefore, Monticello believes that this proposed Frequency is acceptable and no changes to the ITS submittal are necessary.

Date Created: 09/20/2005 09:42 AM by Pete Hearn

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NRC ITS TRACKING

NRC Reviewer

ID	200509261551 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.8; ITS Number: 3.8.1; DOC Number: None; JFD Number: 4; Bases JFD Number: None; Page Number(s): 26
Comment	**NUREG 1433, R3 recommends that the automatic load sequencers be operable. Assuming the time delay relays are equivalent to the automatic load sequencers, please provide justification for not including them in LCO 3.8.1.**
Issue Date	09/26/2005

Close Date	01/23/2006 Resolution requires change to: Other; Docket Response Required? No

Responses

NRC Response by Robert Clark on 01/23/2006	The previous response dated 01/19/2006 addressed the need to verify that the time delay relays are operating within their design limits. Monticello's response to ID No. 200509261748 addresses this issue. However, since Monticello's design does not inclue automatic load sequencers but relies on individual time delay relays, the justification given for not including them in LCO 3.8.1 is acceptable.
Licensee Response by Jerry Jones on 10/06/2005	Improved Standard Technical Specifications (ISTS) LCO 3.8.1.c (Attachment 1, Volume 13, Rev. 0, Page 26 of 294) includes a bracketed requirement that the "Three automatic sequencers" shall be OPERABLE. ISTS 3.8.1 ACTION F (Page 29 of 294) provides the actions when one required automatic load sequencer is inoperable and requires restoration of the required automatic load sequencer to OPERABLE status within 12 hours. The Bases for ISTS ACTIONS F (Page 62 of 294) states that "The sequencer(s) is an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer(s) is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. Therefore, loss of an ESF bus's sequencer affects every major ESF System in the division." Thus, the design described by ISTS LCO 3.8.1.c and included in ACTION F is a single automatic load sequencer that affects and controls all components on the associated division, including the EDG and offsite circuit. That is, if the sequencer is inoperable, nothing in that division will work properly. The Monticello design does not include automatic sequencers that operate all the components associated with an individual ESF bus. As described in the ITS 3.8.1 Background Bases (Page 49 of 294), "Following the trip of offsite power, transfer relays strip nonpermanent loads from the 4.16 kV essential bus. When the EDG is

	tied to the 4.16 kV essential bus, loads are then sequentially connected to its respective 4.16 kV essential bus by individual time delay relays." For the Monticello design, loss of a single individual load timer will impact the starting of only a single component. In addition, failure of the timer may not necessarily impact the EDG or the offsite circuit; only the individual component may be impacted. Therefore, these ISTS requirements were not included in the Monticello ITS, as justified by Justification for Deviations (JFD) 4 (Page 42 of 294). JFD 4 states "The bracketed items specified in ISTS LCO 3.8.1.c and ISTS 3.8.1 ACTION F have been deleted since the Monticello design does not include automatic sequencers. The LCO has been modified and subsequent Conditions and Required Actions have been renumbered, as applicable." The transfer relays and individual time delay relays are tested as required by ITS SR 3.8.1.8 and ITS SR 3.8.1.12. ITS SR 3.8.1.8 (Page 35 of 294) requires verification that on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, permanently connected loads remain energized from the offsite power system and emergency loads are auto-connected through the time delay relays from the offsite power system. ITS SR 3.8.1.12 (Page 40 of 294), in part, requires de-energization of emergency buses, load shedding from emergency buses, and verification that EDG auto-starts from standby condition and energizes auto-connected emergency loads through time delay relays. ITS SR 3.8.1.8 and ITS SR 3.8.1.12 both require the transfer relays and individual time delay relays to support the OPERABILITY requirements for both the required offsite circuits and EDGs, respectively. Thus, if any of the transfer relays or individual time delay relays are inoperable such that they affect offsite circuit or EDG OPERABILITY, the associated required offsite circuit(s) and the associated EDG must be declared inoperable and associated ACTIONS of ITS 3.8.1 entered. Furthermore, the deletion of the LCO and ACTION requirements is consistent with the more recently approved ITS conversions for plants with a design (i.e., the use of individual time delay relays to start components) similar to Monticello (DC Cook Units 1 and 2 and James A. Fitzpatrick Nuclear Power Plant).
NRC Response by Robert Clark on 01/19/2006	ITS SR 3.8.1.8 and ITS 3.8.1.12 only verify that the time delay relays are functional, it does not verify that the timers are operating within their design limits. It is irrelevant if the load sequencer is a single unit that controls the sequencing of all ESF equipment or whether it is done individually with time delay relays. SR 3.8.1.18 specified in NUREG 1433 R3 should be implemented.

Date Created: 09/26/2005 03:51 PM by Robert Clark
Last Modified: 01/23/2006 05:34 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200509261645 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.8 ITS Number: 3.8.1 DOC Number: A.5 JFD Number: 3 Bases JFD Number: None Page Number(s): 30
Comment	**Please comfirm that the accident loads do not exceed the EDG continuous ratings as specified in SR 3.8.1.3. In addition, please provide justification for no power factor requirements.**
Issue Date	09/26/2005

Close Date	03/16/2006 Resolution requires change to: Other Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/06/2005	The ratings of the EDGs are specified in the ITS 3.8.1 Bases Background (Attachment 1, Volume 13, Rev. 0, Page 49 of 294), with the continuous rating of each EDG specified as 2500 kW. ITS SR 3.8.1.3 (Page 30 of 294) requires verification that each EDG is synchronized and loaded and operates for greater than or equal to 60 minutes at a load greater than or equal to 2250 kW and less than or equal to 2500 kW. The Design Basis Accident Loss of Coolant Accident (which includes a loss of offsite power) loads do not exceed the continuous rating of the EDGs (i.e., 2500 kW). The second part of the NRC comment requests a justification for no power factor requirements. ITS SR 3.8.1.7 (ISTS SR 3.8.1.9) (Page 32 of 294) and ITS SR 3.8.1.9 (ISTS SR 3.8.1.14) (Page 37 of 294) include power factor requirements. The bracketed ISTS value for the power factor (0.90) has been replaced with the Monticello plant specific power factor value of 0.95. Therefore, since no power factor requirements have been deleted, a justification does not appear to be required. Note that while ISTS SR 3.8.1.10 (Page 33 of 294) includes a power factor requirement, the entire SR has not been included in the Monticello ITS as described in Justification for Deviations (JFD) 8.
NRC Response by Robert Clark on 12/20/2005	Please verify that the DG power factor specified in Note 2 for ITS 3.8.1.7 and Note 3 for SR 3.8.1.9 are the calculated worst case loading power factors. In addition, the staff noted that the bases implied that the grid voltage can be varied by adjusting the DG field excitation when operating in parallel with the grid. The grid voltage is primarily controlled by the transmission system operators and the automatic voltage regulators installed on the large generating units tied to the grid. The DG excitation should have little or no effect on the grid. However, the staff does believe that potential high grid voltage may prevent the DG from

	obtaining the PF limit specified in the TS due to excessive excitation, and under these conditions Notes 2 and 3 are warrant. Please provide analysis or operating data to demonstrate that the grid voltage can be varied by adjusting the field excitation on a 2.5 Mw DG operating in parallel with the grid. Otherwise, revise the bases for SR 3.8.1.7 and SR 3.8.1.9 accordingly.
Licensee Response by Jerry Jones on 02/24/2006	Based on the NRC reviewer's question, NMC re-evaluated the worst case power factor and determined that it is 0.85 for the Division 1 emergency diesel generator (EDG) and 0.88 for the Division 2 EDG. Improved Standard Technical Specifications (ISTS) SR 3.8.1.9 (Improved Technical Specification (ITS) SR 3.8.1.7) (Attachment 1, Volume 13, Rev. 0, Page 32 of 294 requires the load rejection of the single largest post-accident load for each EDG while ISTS SR 3.8.1.14 (ITS SR 3.8.1.9) requires a load test for each EDG. The load test is performed at two different load ranges; part a is performed between 105% and 110% of the continuous rating and part b is performed between 90% and 100% of the continuous rating. Note 2 of ISTS SR 3.8.1.9 and Note 3 ISTS SR 3.8.1.14 require the testing to be performed at a specific power factor value. Note 2 to ISTS SR 3.8.1.9 (Note 2 to ITS SR 3.8.1.7) has been modified to only require testing "within the power factor limit" and Note 3 to ISTS SR 3.8.1.14 (ITS SR 3.8.1.8) has been modified to only require testing "within the power factor limit" during the load ranges required in part b of the SR (i.e., 90% to 100% of the continuous rating). Currently, neither ISTS SR 3.8.1.9 nor ISTS SR 3.8.1.14 are required in the CTS, and there are no power factor limit requirements in the CTS. The ITS will include the requirement to test at a power factor limit (Note 2 to SR 3.8.1.7 and Note 3 to SR 3.8.1.9), but it will not specify a specific power factor value (e.g., 0.9) in the Notes to the Surveillances. The specific power factor value (0.85 for the Division 1 EDG and 0.88 for the Division 2 EDG) will be included in the ITS Bases for the two Surveillances and will therefore be controlled under ITS 5.5.9, the Technical Specifications Bases Control Program. This program provides for the evaluation of changes to ensure the Bases are properly controlled. Several plants that have converted to the ITS did not include a specific power factor value in the ITS. This type of change was previously approved in the ITS conversion for James A. FitzPatrick Nuclear Power Plant, Quad Cities Unit 1 and 2, Dresden Units 1 and 2, and LaSalle Units 1 and 2. In addition, testing at a power factor limit will not be included for part a of ITS SR 3.8.1.9. During EDG testing at a load equivalent to 105% to 110% of the EDG continuous rating the power factor limit does not have to be met since this part of the test is testing the EDGs at a load in excess of that assumed in the accident analysis (Both the short term and steady state loads during a loss of coolant accident with a loss of offsite power are less than 90% of the continuous rating of the EDGs). All proposed changes to the ITS submittal to reflect the above described changes are shown in the attachment to this response. In addition, the NRC reviewer's concern about the ISTS Bases statement for ISTS SR 3.8.1.9 (Page 70 of 294) and ISTS SR 3.8.1.14 (Page 78 of 294) concerning how adjusting the field excitation on a 2500 kW EDG could affect grid voltage has been re-evaluated. Monticello agrees that this adjustment has little affect and has modified the ITS Bases to better describe why the power factor limit cannot be obtained under certain conditions when the EDG is paralleled to the offsite sources. Changes to the ITS Bases are also shown in the attachment to this response.
Licensee Response by Jerry Jones on 03/13/2006	Based on the NRC reviewer's comment on the second response to RAI 200509261645, NMC has agreed to modify the ISTS Bases markup by deleting INSERT 5 (Attachment 1 Volume 13, Rev. 0 Pages 70 of 294) and all of the text associated with INSERT 6 except the last sentence. Also a typo was corrected in JFD 20 (ITS SR 3.8.1.8 has been changed to ITS SR 3.8.1.9). Changes to the ITS

	Bases and JFDs are also shown in the attachment to this response.

Date Created: 09/26/2005 04:45 PM by Robert Clark
Last Modified: 03/16/2006 01:01 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200509261733 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.8 ITS Number: 3.8.1 DOC Number: None JFD Number: None Bases JFD Number: 7 Page Number(s): 82
Comment	One of the major reasons for performing ITS SR 3.8.1.12 is to demonstrate that the DG is capable of handling the high reactive loads during load sequencing. Please confirm that the loads with the highest starting currents are energized during this test and that they bound the DG response for all other loads in the LOCA sequence. For loads not energized by this test, please confirm that appropriate sequential and overlap testing procedures are provided to verify load sequencing. The bases for ITS SR 3.8.1.14 should clarify that sequential and overlap testing should only be used if during testing there is a potential for equipment damage, undesirable transients, or if testing is not practical due to operating restrictions. However the bases should state that these restrictions are not applicable to loads with the highest starting currents.
Issue Date	09/26/2005

Close Date	01/19/2006 Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/26/2005	Improved Technical Specifications (ITS) SR 3.8.1.12 (Attachment 1, Volume 13, Rev. 0, Page 40 of 294) requires the verification that, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, the emergency buses de-energize, loads are shed from emergency buses; and emergency diesel generator (EDG) auto-starts from standby condition and energizes permanently connected loads in less than or equal to 10 seconds; energizes auto-connected emergency loads through time delay relays; achieves steady state voltage greater than or equal to 3975 V and less than or equal to 4400 V; achieves steady state frequency greater than or equal to 58.8 Hz and less than or equal to 61.2 Hz; and supplies permanently connected and auto-connected emergency loads for greater than or equal to 5 minutes. This Surveillance Requirement is consistent with Current Technical Specification (CTS) 4.9.B.3.a.2) (Page 7 of 294), which requires performance of a similar test. The plant procedures that perform CTS 4.9.B.3.a.2) require the major, automatically-started Loss of Coolant Accident (LOCA) loads (i.e., Core Spray (CS) System. Residual Heat Removal (RHR) System, EDG-Emergency Service Water (ESW), and ESW pumps) to be started and loaded onto the associated EDG. The allowance in the Improved Standard Technical Specifications (ISTS) SR 3.8.1.19 (ITS SR 3.8.1.12) Bases (Page 82 of 294) to

	not start the actual loads is not used in the current plant procedures. Monticello does not intend to change the manner in which the pumps are tested when the ITS is adopted; the CS, RHR, EDG-ESW, and ESW pumps will continue to be started and loaded to meet ITS SR 3.8.1.12. Furthermore, the Monticello ITS SR 3.8.1.12 Bases already states that the allowance to use sequential and overlapping testing in lieu of actually starting and loading the auto-connected loads is only necessary when undue hardship or the potential for undesired operation exists. Therefore, the Bases words proposed by the NRC reviewer are not necessary and do not need to be added to the Monticello ITS.

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NRC ITS TRACKING

NRC Reviewer

ID	**200509261748** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.8** DOC Number: **None** JFD Number: **12** Page Number(s): **43** ITS Number: **3.8.1** Bases JFD Number: **None**
Comment	**ISTS 3.8.1.18 was deleted per JFD 43 because load timers are verified as part of ITS SR 3.8.1.8 and ITS SR 3.8.1.12 c.2. These SRs are functional tests and do not verify the interval between sequenced loads. Please provide justification for no surveillance requirements to verify that the interval between each sequenced load block is within 10% of design.**
Issue Date	**09/26/2005**

Close Date	**02/09/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

NRC Response by Robert Clark on 01/20/2006	To avoid confusion with what is meant by "minimum design load interval," SR 3.8.1.13 should be revised to read, "Verify interval between each sequenced load block is greater than or equal to the design load interval." In addition, please provide justification (operating experience or analysis) as to why the 24 month surveillance frequency is sufficent to ensure that the time delay relays will operate as designed.
Licensee Response by Jerry Jones on 11/03/2005	Improved Standard Technical Specifications (ISTS) SR 3.8.1.18 (Attachment 1, Volume 13, Rev. 0, Page 39 of 294) requires the verification that the interval between each sequenced load block is within +/- [10% of design interval][for each load sequencer timer]. This Surveillance Requirement was not included in the Monticello Improved Technical Specifications (ITS) submittal as justified in Justification for Deviations (JFD) 12 (Page 43 of 294). JFD 12 states "ISTS SR 3.8.1.18 has not been included in the Monticello ITS since the load timers are verified as part of ISTS SR 3.8.1.12.e (ITS SR 3.8.1.8) and ISTS SR 3.8.1.19.c.2 (ITS SR 3.8.1.12.c.2). Subsequent Surveillances have been renumbered, as applicable." ITS SR 3.8.1.8 (Page 35 of 294) requires the verification that on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, permanently connected loads remain energized from the offsite power system and emergency loads are auto-connected through the time delay relays from the offsite power system. ITS SR 3.8.1.12 (Page 40 of 294) requires, in part, verification that the emergency diesel generator (EDG) energizes auto-connected emergency loads through time delay relays, achieves steady state voltage and

	frequency, and supplies the auto-connected loads for 5 minutes. Currently, plant procedures that perform Current Technical Specifications (CTS) 4.9.B.3.a.2) (Page 7 of 294), the Surveillance Requirement equivalent to ITS SR 3.8.1.12, include requirements to verify the interval between load blocks is satisfactory as part of meeting CTS 4.9.B.3.a.2). Thus, if the load intervals are not within the limits specified in the plant procedures, the Surveillance would be considered unacceptable and appropriate Technical Specification actions would be taken. Furthermore, while ITS SR 3.8.1.8 is a new Surveillance Requirement (it is not currently required in the CTS), it is currently performed in plant procedures, and the plant procedures include similar load interval requirements. Thus, the Monticello ITS submittal did not include ISTS SR 3.8.1.18 since the intervals are tested during the performance of CTS 4.9.B.3.a.2), and since ITS SR 3.8.1.12 is equivalent, it was believed that a new Surveillance, ISTS SR 3.8.1.18, was not necessary. Based on the NRC reviewer's concern, Monticello has re-evaluated this position since it is now recognized that the non-inclusion of the ISTS SR 3.8.1.18 in the Monticello ITS submittal was based mainly on presentation preference. To more closely match the ISTS presentation of the load interval testing requirements, Monticello will add ISTS SR 3.8.1.18 into the Monticello ITS. However, the new Surveillance will not include the specific load interval limits; it will state that each load interval be greater than or equal to the minimum design load interval. This presentation will allow Monticello to maintain control of the actual values of the load interval limits, consistent with the current allowances (i.e., CTS 4.9.B.3.a.2) does not include the specific values of the load interval limits). This new Surveillance will be added as shown in the attachment to this response.
Licensee Response by Jerry Jones on 02/02/2006	As stated in proposed Justification for Deviation (JFD) 12, which is provided in the attachment for the original Monticello response to this question, the design load interval has a 10% load interval tolerance. Thus, there is a minimum design load interval (design load interval minus 10%) and a maximum design load interval (design load interval plus 10%). As discussed during the weekly NRC/Monticello Improved Technical Specification (ITS) status phone conversation on 1/26/06, Monticello personnel reiterated the JFD 12 discussion that only the minimum design load interval affected the OPERABILITY of the emergency diesel generators, and that the maximum design load interval affects the affected loads (e.g., Emergency Core Cooling System pumps), and are already required in ITS 3.3.5.1. Therefore, use of the term "minimum design load interval" is the proper term for ITS SR 3.8.1.13. Monticello reviewed the Bases for ITS SR 3.8.1.13 (provided in the attachment for the original Monticello response to this question) and noted that the Frequency description was not consistent with similar Frequency descriptions for other ITS 3.8.1 Surveillances with Frequencies of 24 months (however, the wording is consistent with the wording provided in the Improved Standard Technical Specification (ISTS) Bases for the Surveillance, except for the deletion of the Regulatory Guide reference). The wording provided in other 24 month Surveillances uses the term "operating experience" to justify the 24 month Frequency. Therefore, the Bases for ITS 3.8.1.13 (Attachment 1, Volume 13, Rev. 0, Page 81 of 294) will be modified to be consistent with the wording in other Surveillances performed at a 24 month Frequency. This is shown in the attachment to this response, and this attachment supersedes the Bases page (Page 81 of 294) in the previous attachment.

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NRC ITS TRACKING

NRC Reviewer

ID	**200509261805** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.8** ITS Number: **3.8.1** DOC Number: **None** JFD Number: **11** Bases JFD Number: **None** Page Number(s): **43**
Comment	**ISTS SR 3.8.1.17, is used to verify that with a DG operating in the test mode and connected to its bus, an actual or simulated ECCS signal overrides the test mode by returning the DG to "ready to load" and energizes the emergency loads. However, JFD 11 deletes this surveillance because the EDGs would stay connected to the emergency bus. However, ITS SR 3.8.1.8 states that the emergency loads are auto-connected through time delay relays from the offsite power system. Please clarify that it is acceptable for the EDGs to stay connected to the emergency bus while the loads are being sequenced by offsite power.**
Issue Date	**09/26/2005**

Close Date	**01/20/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/18/2005	Improved Standard Technical Specifications (ISTS) SR 3.8.1.17 (Attachment 1, Volume 13, Rev. 0, Page 39 of 294) requires the verification that with a DG operating in the test mode and connected to its bus, an actual or simulated ECCS signal overrides the test mode by returning the DG to "ready to load" and energizes the emergency loads. This Surveillance Requirement was not included in the Monticello ITS submittal and was deleted as described in Justification for Deviations (JFD) 11 (Page 43 of 294). JFD 11 states "ISTS SR 3.8.1.17 is not included in the Monticello ITS since this feature was not included in the Monticello design. This SR demonstrates that with an EDG operating in the test mode and connected to its bus, an ECCS initiation signal overrides the test mode and returns the EDG to ready-to-load operation. At Monticello, with an EDG connected to its bus, if an ECCS initiation signal were received, the EDG would stay connected to its bus. Furthermore, the EDGs do not perform any safety-related function for a LOCA event (e.g., ECCS initiation) since the offsite circuits remain available. Therefore, this SR is not applicable." Improved Technical Specifications (ITS) SR 3.8.1.8 (Page 35 of 294) requires the verification that on an actual or simulated Emergency Core Cooling System (ECCS) initiation signal, permanently connected loads remain energized from the offsite power system and emergency loads are auto-connected through the time delay relays from the

	offsite power system. Thus, ITS SR 3.8.1.8 does not require the emergency diesel generator (EDG) to be tested. However, the test will normally be performed with the EDG in the standby condition, consistent with the ISTS requirements; i.e., it will not be connected to the associated bus prior to the commencement of the SR. Furthermore, since the Monticello design does not disconnect the EDG from the associated bus when being tested, if an ECCS initiation signal is received the EDG is considered inoperable when it is connected to its associated bus and paralleled with the grid during testing (e.g., during performance of ITS SR 3.8.1.3 (Page 30 of 294)). Monticello declares the EDG inoperable when paralleled with the grid because if a grid failure were to occur in this condition, the EDG could be overloaded followed by the protective relaying tripping the EDG output breaker, thus resulting in an EDG overspeed trip. Due to this EDG overspeed trip potential when the EDG is paralleled to the grid, manually placing the EDG in a ready to load condition (following an overspeed trip) is proceduralized in the EDG Surveillances. In addition, during the development of the Monticello response to this question, it was noted that ITS SR 3.8.1.8 Note 1 provides an allowance for the EDG start portion of the SR. However, as described in JFD 18, the EDG start requirements are not required as part of ITS SR 3.8.1.8. Therefore, Note 1 will be deleted and its deletion is covered by the wording of JFD 18. Appropriate ITS Bases changes will also be made. The proposed ISTS Markup and ISTS Bases Markup page changes (Pages 35 of 294 and 75 of 294) are provided in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	**200509271127** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	**ITS Section:** 3.8 **ITS Number:** 3.8.1 **DOC Number:** None **JFD Number:** 15 **Bases JFD Number:** None **Page Number(s):** 44
Comment	**ISTS SR 3.8.1.4 recommends that the fuel oil volume for the day and base tank be check every 31 days. However, this surveillance requirement is not included in the proposed ITS conversion. The licensee states in part that; the fuel oil in the day tank and the base tank (i.e.,the engine mounted tank)is not necessary to meet the 7 day fuel oil requirement. Only the fuel oil in the common storage tank is used to meet the 7 day fuel oil requirement. This 7 day limit is verified in ISTS SR 3.8.3.1 (ITS SR 3.8.3.1). The fuel oil transfer system includes two pumps that are capable of transferring fuel oil from the common storage tank to each day tank, and two pumps per EDG that are capable of transferring fuel oil from the associated day tank to the associated base tank. Please clarify if the fuel transfer pumps for both the day and base tanks will automatically start on low fuel oil level or must the pumps be manually started by the operator. If manually started, is a low level alarm provided in the main control room?**
Issue Date	**09/27/2005**

Close Date	**01/23/2006**	Resolution requires change to: Other Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/12/2005	Improved Standard Technical Specifications (ISTS) SR 3.8.1.4 (Attachment 1, Volume 13, Rev. 0, Page 30 of 294) states "Verify each day tank [and engine mounted tank contains[s] greater than or equal to [900] gal of fuel oil." This Surveillance Requirement was not included in the Monticello Improved Technical Specifications (ITS) submittal as justified in Justification for Deviations (JFD) 15 (Page 44 of 294). JFD 15 states "ISTS SR 3.8.1.4 requires verification that the fuel oil level in the day tank and engine mounted tank is within a specified limit. This Surveillance has not been adopted in the Monticello ITS. At Monticello, the fuel oil in the day tank and the base tank (i.e., the engine mounted tank) is not necessary to meet the 7 day fuel oil requirement. Only the fuel oil in the common storage tank is used to meet the 7 day fuel oil requirement. This 7 day limit is verified in ISTS SR 3.8.3.1 (ITS SR 3.8.3.1). The fuel oil transfer system includes two pumps that are capable of transferring fuel oil from the common storage tank to each day tank, and two pumps per EDG that are capable of transferring fuel oil from the associated day tank to the associated base tank. ISTS SR 3.8.1.6 (ITS SR 3.8.1.5) verifies that the fuel oil transfer system can operate as designed at the

	same Frequency as ISTS SR 3.8.1.4. Provided the fuel oil transfer pumps are properly operating, the fuel oil level in each day tank and base tank will be adequately maintained to support EDG OPERABILITY. In addition, an alarm is provided to alert the operator to a problem with the fuel oil transfer pumps associated with the common storage tank. Therefore, ISTS SR 3.8.1.4 is redundant to the fuel oil transfer pumps Surveillance (ISTS SR 3.8.1.6) and is not necessary to be included in the Monticello ITS. This is also consistent with the current licensing basis, since this Surveillance is not included in the CTS. Subsequent Surveillances have been renumbered, as applicable." The day tanks, base tanks, common fuel oil storage tank, and fuel oil transfer system is discussed in the ITS 3.8.1 Bases, Background section, INSERT 3 (Page 50 of 294). The Bases Insert states "Each EDG has its own day tank and base tank. Both EDGs utilize a common fuel oil storage tank. The fuel oil transfer system, which includes a fuel oil transfer pump and a fuel oil service pump, is capable of transferring fuel oil from the fuel oil storage tank to both day tanks. Both the fuel oil transfer pump and the fuel oil service pump are individually capable of maintaining the level in the day tank when both EDGs are operating at full load. The fuel oil transfer system also includes two day tank fuel oil transfer subsystems. Each day tank fuel oil transfer subsystem is capable of automatically transferring fuel oil from the day tank to the associated base tank. Each day tank fuel oil transfer subsystem includes two pumps, and each pump starts automatically on a level signal from one the base tank level switch. One pump starts when the level in the base tank drops below the normal level and the second pump starts when the base tank level drops to the low level." As stated above in the ITS 3.8.1 Bases Insert, the transfer of fuel oil from the day tank to the associated base tank is automatic. Both associated day tank fuel oil pumps have an automatic start signal, based on a low level in the base tank. The fuel oil transfer system that transfers fuel oil from the common fuel oil storage tank to the emergency diesel generator (EDG) day tanks is designed to keep the EDG day tanks continuously full. The fuel oil service pump is operated continuously, supplying fuel oil to both EDG days tanks, and any excess fuel oil is recirculated back to the common fuel oil storage tank by gravity flow from the overflow connection near the top of each day tank. Instrumentation is available to indicate abnormal status of the system/tanks. If flow through an EDG day tank overflow line should stop, or if level in a day tank should drop significantly, annunciators in the control room are actuated. If the alarm is due to the loss of the normally operating fuel oil service pump, the fuel oil transfer pump can be manually started from the control room. No automatic start is provided for the fuel oil transfer pump. In addition, during the development of the Monticello response to this question, a typographical error was noted in ITS 3.8.1 Bases INSERT 3. This will be corrected as shown in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	200510031416 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.4 ITS Number: 3.4.3 DOC Number: LA.2 JFD Number: None Bases JFD Number: None Page Number(s): 53
Comment	CTS SR 4.6.8.1 appears to place all 8 S/RV in the Inservice Testing Program. ITS SR 3.4.3.1 appears only to place 7 "required" S/RV in the Inservice Testing Program. ITS SR 3.4.3.2 also refers to "required" SR/Vs. For purposes of this discussion, assume that Monticello designates one specific valve, called the "8th valve," as "not required." Is the intent of the proposed ITS to allow the 8th valve to be outside of the scope of any maintenance and testing program or any TS for an indefinite time, so long as the valve is "not required?"
Issue Date	10/03/2005

Close Date	10/19/2005 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/12/2005	Current Technical Specifications (CTS) 3.6.E.1 (Attachment 1, Volume 9, Rev. 0, Page 50 of 255) requires seven safety/relief valves (S/RVs) to be OPERABLE and includes a statement that eight S/RVs are set at the same pressure (less than or equal to 1120 psig). CTS 4.6.E.1 (Page 50 of 255) states "Safety/relief valves shall be tested or replaced each refueling interval in accordance with the Inservice Testing Program." This requirement does not state the number of S/RVs that must be tested each refueling interval. Improved Technical Specification (ITS) LCO 3.4.3 (Page 58 of 255) requires the safety function of seven S/RVs to be OPERABLE and ITS SR 3.4.3.1 (Page 59 of 255) requires verification that the safety function lift setpoints of the required S/RVs are 1109 plus or minus 33.2 psig and following testing, lift settings shall be 1109 plus or minus 11.0 psig. The detail that the S/RVs shall be tested or replaced each refueling interval has been relocated to the Inservice Testing Program in accordance with Discussion of Change (DOC) LA.2 (Page 53 of 255). ITS 5.5.5 (Attachment 1, Volume 17, Rev. 0, Page 83 of 143) requires the Inservice Testing Program to be established, implemented, and maintained. The eight installed S/RVs are included in the Monticello Pump and Valve Inservice Testing (IST) Program Plan and there is no intent in the conversion to the ITS to allow the one S/RV (i.e., the "8th" S/RV) to

	be outside of the scope of the maintenance and testing program. The Monticello IST Program will continue testing requirements on all eight of the S/RVs in accordance with the ASME Operations and Maintenance (OM) Code of record as it applies to the Monticello IST fourth 10 year interval. DOC LA.2 is not a deletion of the CTS 4.6.E.1 requirement. CTS 4.6.E.1 requires S/RVs to be tested or replaced each refueling interval. This requirement is being relocated to the Inservice Testing Program (where it currently exists) as described in DOC LA.2. In addition, once relocated, this requirement can only be changed in accordance with 10 CFR 50.55a.

Date Created: 10/03/2005 02:16 PM by David Roth
Last Modified: 10/19/2005 07:07 AM

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NRC ITS TRACKING

NRC Reviewer

ID	200510031651 Conference Call Requested? No
Category	Minor Technical
ITS Information	ITS Section: 3.0 ITS Number: 3.0 DOC Number: None JFD Number: None Bases JFD Number: 7 Page Number(s): 43
Comment	By allowing the requirements of Surveillances to be satisifed by unplanned events, even for precluded surveillances in the Mode of occurrance, is not intended to change Tech Spec requirements. We are not intending the use of the word "precluded" to be synonymous with "prohibited." The intent is that if an event can satisfy an SR, even if the SR were not normally perfomed during the Mode in which the event occured, then credit can be given for the performance of teh SR. You can certainly delete the Bases sentence as you propose, however, you may want to reconsider?
Issue Date	10/03/2005

Close Date	12/19/2005	Resolution requires change to: None Bases JFD Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/11/2005	The Improved Standard Technical Specifications (ISTS) SR 3.0.1 Bases (Attachment 1, Volume 5, Rev. 0, Page 43 of 63) states "Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition." The last sentence was not included in the Monticello Improved Technical Specifications (ITS) and was deleted with Justification for Deviations (JFD) 7 (Page 49 of 63). JFD 7 states "The ITS SR 3.0.3 Bases allows credit to be taken for unplanned events that satisfy Surveillances. The Bases further states that this allowance also includes those SRs whose performance is normally precluded in a given MODE or other specified condition. This portion of the allowance has been deleted. As documented in Part 9900 of the NRC Inspection Manual, Technical Guidance - Licensee Technical Specifications Interpretations, and in the Bases Control Program (ITS 5.5.10), neither the Technical Specifications Bases nor Licensee generated interpretations can be used to change the Technical Specification requirements. Thus, if the Technical Specifications preclude performance of an SR in certain MODES (as is the case for some SRs in ITS Section 3.8), the Bases cannot change the Technical Specifications requirement and allow the SR to be credited for being performed in the restricted MODES, even if the performance is

	unplanned." Based on the NRC reviewers comment, NMC has re-reviewed the proposed deletion in the Monticello ITS and has concluded the allowance is not necessary. The deleted sentence is basically providing an example of a type of SR for which credit could be taken during an unplanned event. If the sentence is not included, then a "precluded" SR (for example, an SR that cannot normally be performed in MODE 1 because it could cause a trip of the unit) could still be credited during an unplanned event. That is, deletion of the sentence does not result in Monticello not being able to credit an unplanned event in MODE 1 from meeting an SR that cannot normally be performed in MODE 1. The allowance still exists as specified in the first two sentences. Monticello deleted the last sentence to prevent confusion and misunderstanding as to what "precluded" actually means. Furthermore, "prohibited" SRs, like those in ITS 3.8.1 (those that have Notes that say they cannot be performed in certain MODES), are not allowed to use the SR 3.0.1 allowance, unless specifically stated in the applicable Notes to the ITS 3.8.1 SRs. Certain Surveillances in ITS 3.8.1 (SR 3.8.1.6, SR 3.8.1.7, SR 3.8.1.8, SR 3.8.1.9, SR 3.8.1.11, and SR 3.8.1.12) (Pages 31, 32, 35, 37, 38, and 40 of 294) and ITS 3.8.4 (SR 3.8.4.3) (Page 153 of 294) include a Note that restricts the normal performance of the associated Surveillance in specific MODES. These are the only Notes in the Monticello ITS that restricts SRs from being performed in specific MODES or conditions. However, the same Note also includes the following statement, "Credit may be taken for unplanned events that satisfy this SR." Therefore, the deleted phrase in the ISTS SR 3.0.1 Bases is not necessary for these types of SRs. In addition, during the development of the Monticello response to this question, it was noted that JFD 7 incorrectly stated it was discussing the ITS SR 3.0.3 Bases. It should have stated it was discussing the ITS SR 3.0.1 Bases. This will be corrected as shown in the attachment to this response.

Date Created: 10/03/2005 04:51 PM by bob tjader
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NRC ITS TRACKING

NRC Reviewer

ID	200510051403 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.4 ITS Number: 3.4.5 DOC Number: A.5 JFD Number: 10 Bases JFD Number: None Page Number(s): 92
Comment	Issue is addition of drywell equipment drain sump monitoring system being used, via overflow, to monitor floor drain sump. JFD 10 says this is part of CLB based on amendment 137 (ADAMS ML031980275) and the TS Bases. Amend. 137's safety evaluation stated on page 8, "The system becomes inoperable during periods when the floor drain sump level and flow indications are not capable of being monitored. Once the drywell floor drain sump is overflowing to the equipment drain sump, NMC can use the drywell equipment drain sump monitoring system to quantify leakage (i.e., unidentified leakage) into the floor drain sump." If the floor drain sump monitoring system is inoperable, what TS Required Action does Monticello propose being in from the time "the system becomes inoperable" until "the drywell floor drain sump is overflowing to the equipment drain?" grammar corrected shortly after posting 200510051405
Issue Date	10/05/2005

Close Date	10/19/2005 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/11/2005	The Monticello Improved Technical Specifications (ITS) LCO 3.4.5.a (Attachment 1, Volume 9, Rev. 0, Page 96 of 255) requires "Either the drywell floor drain sump monitoring system or the drywell equipment drain sump monitoring system with the drywell floor drain sump overflowing to the equipment drain sump" to be OPERABLE. If the drywell floor drain sump monitoring system is inoperable and the drywell floor drain sump is not overflowing to the equipment drain sump, then LCO 3.4.5.a is not being met. In this case, ITS 3.4.5 Condition A (Page 96 of 255) must be entered, since the Condition A states "LCO 3.4.5.a not met", and the action of Required Action A.1, satisfy the requirements of LCO 3.4.5.a within 30 days, must be taken. Once the drywell floor drain sump begins to overflow to the equipment drain sump and, provided all the required instrumentation associated with the drywell equipment drain sump monitoring system is OPERABLE, then LCO 3.4.5.a is met and ITS 3.4.5 ACTION A can be exited.

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NRC ITS TRACKING

NRC Reviewer

ID	**200510141334** Conference Call Requested? **No**
Category	**Minor Technical**
ITS Information	ITS Section: **3.1** ITS Number: **3.1.7** DOC Number: **M.1** JFD Number: **2** Bases JFD Number: **None** Page Number(s): **181**
Comment	**Second Completion Times (CT) have been changed to 14 days from 10 days in the STS, reflecting the retention of the existing CT of 7 days to restore boron concentration. Recommend adopting the recently approved TSTF-439, eliminating the second CT, and adding an example and dsicussion to section 1.**
Issue Date	**10/14/2005**

Close Date	**12/21/2005** Resolution requires change to: `Typed ITS` Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/05/2006	Technical Specification Task Force (TSTF) - 439 was not approved at the time of the Monticello Improved Technical Specifications (ITS) submittal, thus it was not included in the Monticello ITS submittal. However, Monticello has since learned that the NRC has agreed to incorporate TSTF-439, Rev. 2 into Revision 3.1 of NUREG-1433, the BWR/4 Improved Standard Technical Specifications (ISTS). This is documented in the NRC-issued revision 3.1 of NUREG-1433 (the changes proposed in TSTF-439 are incorporated into this new revision). Therefore, Monticello will adopt TSTF-439, Rev. 2. The proposed changes to the Monticello ITS are shown in the attachment to this response. It should be noted that the TSTF-439 changes not only affect ITS 3.1.7, but also ITS 1.0, ITS 3.8.1, and ITS 3.8.7. The proposed changes will be provided as attachments. The attachment to this specific response includes the ITS 3.1.7 changes. Additional responses will be added to include the remaining changes for each affected ITS Specification (ITS 1.0, ITS 3.8.1, and ITS 3.8.7).
Licensee Response by Jerry Jones on 01/05/2006	The proposed changes to ITS 1.0 (due to the adoption of TSTF-439) are provided as an attachment to this response.
Licensee Response by Jerry Jones on 01/05/2006	The proposed changes to ITS 3.8.7 (due to the adoption of TSTF-439) are provided as an attachment to this response.
Licensee Response by Jerry Jones on 01/05/2006	The proposed changes to ITS 3.8.1 (due to the adoption of TSTF-439) are provided as an attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	200510141347 Conference Call Requested? No
Category	Minor Technical
ITS Information	ITS Section: 3.1 ITS Number: 3.1.7 DOC Number: None JFD Number: 3 Bases JFD Number: None Page Number(s): 181
Comment	The LCO requirement is for boron concentration to be within limits, both in the STS and in the CTS. The Condition has been changed to reflect sodium pentaborate concentration not within limits. Recommend retaining the Condition A and associated Required Actions that address Boron concentration; the sodium pentaborate concentration is an indicator whether the Boron concentration is met.
Issue Date	10/14/2005

Close Date	12/19/2005 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 10/20/2005	Current Technical Specifications (CTS) 3.4.B.1 (Attachment 1, Volume 6, Rev. 0, Page 169 of 231) requires the standby liquid control (SLC) tank to contain a boron bearing solution of liquid that satisfies the volume, concentration, and enrichment requirements of Figure 3.4-1. CTS Figure 3.4-1 (Page 171 of 231) is a plot of Indicated Tank Volume versus Weight Percent Sodium Pentaborate Concentration, with the chemical formula of sodium pentaborate being provided. CTS 3.4.B.2 (Page 169 of 231) requires temperature to be within the limits of Figure 3.4-2. CTS Figure 3.4-2 (Page 172 of 231) is a plot of Measured Solution Temperature versus Weight Percent Sodium Pentaborate in Solution, again with the chemical formula of sodium pentaborate being provided. Thus, the CTS requirement reflects sodium pentaborate concentration, not boron concentration. The Improved Technical Specification (ITS) has been written to retain the term "sodium pentaborate" in lieu of the Improved Standard Technical Specifications (ISTS) generic term of "boron concentration." ITS SR 3.1.7.1 (Page 182 of 231) requires the verification that the available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1. ITS Figure 3.1.7-1 (Page 185 of 231) is a plot of Sodium Pentaborate Concentration vs. Indicated Tank Volume. ITS SR 3.1.7.2 (Page 182 of 231) requires verification that temperature of sodium pentaborate solution is within limits of Figure 3.1.7-2. ITS Figure 3.1.7-2 (Page 187 of 231) is a plot of Measured Solution Temperature versus Sodium Pentaborate In Solution. It should be noted that the ISTS SR 3.1.7.1 and SR 3.1.7.2 term is also "sodium pentaborate," not "boron concentration."

	Furthermore, the term used in ITS 3.1.7 ACTION A is also sodium pentaborate, consistent with the CTS requirements. These changes (use of the term "sodium pentaborate" in lieu of "boron concentration") were justified in Justification for Deviations (JFD) 3 (Page 189 of 231). JFD 3 states "The proper Monticello nomenclature has been used (CTS Figures 3.4.-1 and 3.4-2). This is also consistent with the nomenclature used in SR 3.1.7.1 and SR 3.1.7.2." Monticello could have also justified the changes by saying that the terms were changed for consistency with similar terms in this Specification. Monticello's position is that the same term for the solution description should be used, and since the CTS term is "sodium pentaborate," and the same term is also used in some of the requirements in ISTS 3.1.7, this is the term selected for the Monticello ITS. Furthermore, it is more descriptive of the type of boron being used in the SLC System.

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NRC ITS TRACKING

NRC Reviewer

ID	200510141527 Conference Call Requested? No
Category	**Minor Technical**
ITS Information	ITS Section: **3.1** DOC Number: **None** JFD Number: **5** Page Number(s): **183** ITS Number: **3.1.7** Bases JFD Number: **None**
Comment	**The SR 3.1.7.9 frequency has been changed to reflect that Monticello does not have temperature indication on the SLC piping, and instead uses room temperature as an indicator of potential line blockage through sodium pentaborate solution solidification. Temperature indication on the SLC piping is also an indicator of the operability of the heat tracing; which room temperature indication would not be able to discern. If room temperature is low enough there may be line blockage if either the heat tracing were not operating or if a porting of the system was not kept hot by the heat tracing. Recommend deleting the note.**
Issue Date	**10/14/2005**

Close Date	**12/19/2005** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/20/2005	Improved Standard Technical Specifications (ISTS) SR 3.1.7.9 (Improved Technical Specifications (ITS) SR 3.7.1.9) (Attachment 1, Volume 6, Rev. 0, Page 183 of 231) requires verification that all heat traced piping between the storage tank and pump suction is unblocked. The second Frequency of ISTS SR 3.1.7.9 is "Once within 24 hours after solution temperature is restored within the limits of [Figure 3.1.7-2]." This Frequency has been modified in ITS SR 3.1.7.9 to be "Once within 24 hours after room temperature in the vicinity of the SLC pumps is restored within the solution temperature limits of Figure 3.1.7-2." Furthermore, a Note is included that states this Frequency is only required if SLC pump suction lines heat tracing is inoperable. ISTS SR 3.1.7.3 (Page 182 of 231) requires verification that the temperature of the pump suction piping is within the limits of Figure 3.1.7-2. This Surveillance has been modified in the Monticello ITS because the plant does not include temperature indication of the pump suction piping. ITS SR 3.1.7.3 (Page 182 of 231) requires verification that either the temperature of the room in the vicinity of the standby liquid control (SLC) pumps is within the solution temperature limits of Figure 3.1.7-2 or the SLC pump suction lines heat tracing is OPERABLE. This Surveillance is consistent with Current Technical Specifications (CTS) 4.4.B.3.c and CTS 3.4.B.3 (Page 169 of 231). The purpose of this Surveillance is to help ensure that the proper

	borated solution temperature of the pump suction piping is maintained. Maintaining a minimum specified room temperature is important in ensuring that the boron remains in solution and does not precipitate out in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 5 degrees F margin will be maintained above the saturation temperature. Thus, as long as the room temperature is within the limits of Figure 3.1.7-2, the boron solution that is in the suction piping will remain in solution. An acceptable alternate requirement is to verify the pump suction lines heat tracing is OPERABLE, since the heat tracing design is to maintain proper suction piping temperature. The modifications to the second Frequency in ITS SR 3.1.7.9 were made to reflect the changes in ITS SR 3.1.7.3. Three piping sections of the SLC System are provided with heat tracing and are insulated. These sections are from the storage tank outlet to the test tank outlet, from the test tank outlet to the suction of each pump, and the pump plunger casings. Therefore, all the piping from the storage tank to the pump suction is completely heat traced. If the heat tracing is OPERABLE, precipitation is not expected to occur on the pump suction piping due to the capacity of the heat tracing, and therefore there is no reason to perform ITS SR 3.1.7.9. The Note is necessary since it clarifies that ITS SR 3.1.7.9 is not required to be performed at the accelerated Frequency after the temperature in the vicinity of SLC pumps was restored to within the solution temperature limits of Figure 3.1.7-2, as long as the SLC pump suction lines heat tracing is OPERABLE.

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NRC ITS TRACKING

NRC Reviewer

ID	**200510171109** Conference Call Requested? **No**
Category	**Minor Technical**
ITS Information	ITS Section: **4.0** ITS Number: **4.0** DOC Number: **None** JFD Number: **1** Bases JFD Number: **None** Page Number(s): **13**
Comment	**Section 4.3.1.1 and 4.3.1.2 Justify the deletion of the "average U-235 enrichment of [4.5]weight percentage".**
Issue Date	**10/17/2005**

Close Date	**01/18/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/20/2005	Improved Standard Technical Specifications (ISTS) 4.3.1.1 (Attachment 1, Volume 16, Rev. 0, Page 13 of 19) states "The spent fuel storage racks are designed and shall be maintained with:" and ISTS 4.3.1.1.a states "Fuel assemblies having a maximum [k-infinity of [1.31] in the normal reactor core configuration at cold conditions] [average U 235 enrichment of [4.5] weight percent]." Thus, ISTS 4.3.1.1 provides two options for the spent fuel storage racks, either a k-infinity value or a U-235 enrichment value. In the Monticello Improved Technical Specifications (ITS), the k-infinity option was provided in ITS 4.3.1.1.a (i.e., "Fuel assemblies having a maximum k-infinity of 1.33 in the normal reactor core configuration at cold conditions"). This option is consistent with Current Technical Specifications (CTS) 5.5.B (Page 6 of 19), which only specifies the k-infinity value for the spent fuel storage racks. ISTS 4.3.1.2 (Page 13 of 19) states "The new fuel storage racks are designed and shall be maintained with:" and ISTS 4.3.1.2.a (Page 15 of 19) states "Fuel assemblies having a maximum [k-infinity of [1.31] in the normal reactor core configuration at cold conditions] [average U 235 enrichment of [4.5] weight percent]." Thus, ISTS 4.3.1.2 provides two options for the new fuel storage racks, either a k-infinity value or a U-235 enrichment value. In the Monticello Improved Technical Specifications (ITS), the k-infinity option was provided in ITS 4.3.1.2.a (i.e., "Fuel assemblies having a maximum k-infinity of 1.31 in the normal reactor core configuration at cold conditions"). This option is consistent with Current Technical Specifications (CTS) 5.5.A (Page 6 of 19), which only specifies the k-infinity value for the new fuel storage racks. The detail in both ISTS 4.3.1.1 and ISTS 4.3.1.2 concerning the "average U-235 enrichment of [4.5] weight percent" has been deleted as described in Justification for Deviations (JFD) 1 (Page 17 of

	19), which states "The brackets have been removed and the proper plant specific information/value has been provided." Since the requirement in ITS 4.3.1.1.a is consistent with requirement in CTS 5.5.B and the requirement in ITS 4.3.1.2.a is consistent with the requirements in CTS 5.5.A, and the requirements for "k-infinity" and "U-235 enrichment" in the ISTS are both bracketed, deletion of the U-235 option using JFD 1 is acceptable. The deletion of the U-235 option from the ITS is also consistent with the most recently approved Boiling Water Reactor (BWR) ITS conversion (the James A. FitzPatrick Nuclear Power Plant ITS conversion, amendment 274, issued July 3, 2002).

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NRC ITS TRACKING

NRC Reviewer

ID	**200510171402** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.4** DOC Number: **A.4** JFD Number: **8** Page Number(s): **107** ITS Number: **3.4.5** Bases JFD Number: **None**
Comment	**LCO 3.4.5 states: The following RCS leakage detection instrumentation shall be OPERABLE: a. Either the drywell floor drain sump monitoring system or the drywell equipment drain sump monitoring system with the drywell floor drain sump overflowing to the equipment drain sump; and b. Drywell particulate radioactivity monitoring system Proposed Bases B 3.4.5 Action A.1 state: With LCO 3.4.5.a not met, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage. With LCO 3.4.5.a not met, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the alternate form of leakage detection that is normally still available and the fact that the LEAKAGE is still being determined every 12 hours. ------------------------------ For discussion purposes, assume the following initial conditions: Full power, all equipment operable, 4.81 gpm unidentified leakage - leakage unchanged for 60 days. At time 0, RCS Leakage Detection Instrumentation failure causes Proposed ITS LCO 3.4.5.a. not to be satisfied; it will take 14 days before 3.4.5.a. can be satisfied. Given the above initial conditions, provide more details about what "alternate form of leakage detection" is available to determine leakage every 12 hours per SR 3.4.4.1. 10 CFR 50.36 requires that a technical specification limiting condition for operation of a nuclear reactor be established for installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. State the LCOs for the alternate forms of leak detection. State whether the drywell particulate radioactivity monitoring system required by Proposed ITS 3.4.5.b. would be capable of satisfying ITS SR 3.4.4.1. Specifically, if the "indication of changes in leakage" provided by the rad monitor could detect the 0.2 gpm increase in leakage that would result in exceeding the 5 gpm unidentified leakage. Discuss what actions apply if SR 3.4.4.1 cannot be performed while or because LCO 3.4.5.a is not met.**
Issue Date	**10/17/2005**

Close Date	**11/03/2005** Resolution requires change to: None

	Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/26/2005	Improved Technical Specifications (ITS) LCO 3.4.5.a (Attachment 1, Volume 9, Rev. 0, Page 96 of 255) requires either the drywell floor drain sump monitoring system or the drywell equipment drain sump monitoring system with the drywell floor drain sump overflowing to the equipment drain sump to be OPERABLE. The ITS 3.4.5 Bases (Pages 102, 103, and 104 of 255) describe the various methods of LEAKAGE determination that can be used to monitor unidentified leakage and satisfy the requirements of LCO 3.4.5.a. If the drywell floor drain sump monitoring system is used to meet LCO 3.4.5.a, there are three methods of LEAKAGE determination to satisfy the LCO 3.4.5.a requirement. As stated in the ITS 3.4.5 Bases Background section, INSERT 3 (Page 103 of 255), each method of LEAKAGE determination of the drywell floor drain sump monitoring system (level recorder in the control room, drywell floor sump fill rate, and flow integrator) is sensitive enough to detect leak rate changes better than one gallon per minute in a one hour period and therefore may be used to satisfy the LCO requirement. If the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump monitoring system can be used to quantify unidentified LEAKAGE. As stated in the ITS 3.4.5 Bases Background section, INSERT 3A (Page 104 of 255), the drywell equipment drain sump monitoring system also includes three methods of Leakage determination to satisfy the LCO 3.4.5.a requirement (level recorder in the control room, drywell floor sump fill rate, and flow integrator). The ITS 3.4.5 Bases LCO section (Pages 105 and 106 of 255) states that for each monitoring system, any one of the three methods can be used to meet LCO 3.4.5.a. Thus, if none of the six total methods of monitoring are capable of satisfying LCO 3.4.5.a, ITS 3.4.5 ACTION A (Page 96 of 255) must be entered and a 30 day Completion Time is provided to restore the equipment to service (i.e., restore compliance with LCO 3.4.5.a). The Bases for ITS 3.4.5 ACTION A (Pages 107 and 108 of 255) provide the following justification for the 30 day Completion Time. "With LCO 3.4.5.a not met, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the alternate form of leakage detection that is normally still available and the fact that the LEAKAGE is still being determined every 12 hours." The above Bases paragraph refers to "the alternate form of leakage detection that is normally still available." This alternate form is the other instrument required by LCO 3.4.5.b (Page 96 of 255), the drywell particulate radioactivity monitoring system. However, this alternate form of leakage detection is not the one that is capable of meeting SR 3.4.4.1 (Page 76 of 255), nor does the above paragraph state that it is the method used to meet SR 3.4.4.1. The above Bases paragraph states that there is an alternate form of leakage detection still available, and that LEAKAGE is still being determined every 12 hours (per SR 3.4.4.1). The assumption in the Bases is that there is another method to quantify LEAKAGE in order to meet the requirements of SR 3.4.4.1. While not a method qualified to meet the requirements of LCO 3.4.5, one method that might be used to quantify LEAKAGE during the time the LCO 3.4.5.a monitors are inoperable is to manually start and time the floor drain sump pump run time (until it trips on low level), and using this time, the known pump flow rate, and the time from last operating the pump (which previously was turned off on low level), calculate a leakage rate. Another example (if all six methods allowed by LCO 3.4.5.a are

	inoperable) of a method to quantify LEAKAGE could be by using a different, normally not installed, indicator to measure LEAKAGE. For example, the level recorder is one of the methods allowed to meet LCO 3.4.5.a. In lieu of using the installed level recorder (i.e., it is inoperable), a temporary strip chart recorder could be connected to output of the level transmitter that feeds the normally installed level recorder. This temporary strip chart recorder could then be used to quantify LEAKAGE during the time the LCO 3.4.5.a required monitors are inoperable. These methods (pump run time method and temporary strip chart recorder method), and any other method used to quantify LEAKAGE during the time the LCO 3.4.5.a monitors are inoperable, is not specified in an LCO statement. The only leakage detection instruments that quantify leakage covered by an LCO are those required by ITS 3.4.5.a. If these LCO 3.4.5.a required instruments are inoperable, operation is only allowed for 30 days, as discussed above. As stated in the ITS 3.4.5 Bases Background section (Page 105 of 255), while the drywell particulate radioactivity monitoring system required by ITS LCO 3.4.5.b (Page 96 of 255) is capable of monitoring leakage as low as 1E-9 microcuries/cc, it is not capable of quantifying LEAKAGE rates. Therefore, this system cannot be used to meet SR 3.4.4.1 (Page 76 of 255), which requires verification that required LEAKAGE is within limits (i.e., the LEAKAGE must be quantified to determine if the LEAKAGE limits are met). If it is determined that SR 3.4.4.1 cannot be performed (i.e., the LEAKAGE cannot be quantified), then the plant must enter SR 3.0.3 (Attachment 1, Volume 5, Rev. 0, Page 28 of 63). SR 3.0.3 allows up to 24 hours (which is the greater of 12 hours (the limit of the specified Frequency) and 24 hours) to perform the "missed" Surveillance, and if not performed within this 24-hour period, ITS LCO 3.4.4 must be declared not met and the applicable Condition(s) entered. In addition, during the development of the Monticello response to this question, it was noted that the Bases for ITS SR 3.4.4.1 (Page 83 of 255) incorrectly stated that an alternate method to quantify LEAKAGE is the drywell equipment drain sump monitoring system with the drywell floor drain sump overflowing to the drywell equipment drain sump. This method is a normally allowed method in accordance with ITS LCO 3.4.5.a. The Bases will be revised to state that an alternate method to quantify LEAKAGE is using drywell sump pump run times, which, as described above, is one of the methods that can be used to quantify LEAKAGE if the monitors required by LCO 3.4.5.a are inoperable. This alternate method is consistent with the alternate method provided in the Quad Cities 1 and 2 and Dresden 2 and 3 ITS conversion. This will be corrected as shown in the attachment to this response. Furthermore, in ITS LCO 3.4.5.a, the term "equipment drain sump" will be changed to "drywell equipment drain sump", consistent with the first usage of the term in ITS LCO 3.4.5.a. This will also be corrected as shown in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	200510172130 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.6; ITS Number: 3.6.1.1; DOC Number: L.1; JFD Number: None; Bases JFD Number: None; Page Number(s): 7
Comment	CTS 4.7.A.2.d "The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration." CTS 4.7.A.1.c requires visual inspection of the accessible portions of the suppression chamber interior each refueling interval. These requirement details are not explicitly retained in SR 3.6.1.1.1, or stated in the associated Bases. They are removed detail type changes and should be included in Table R, as an LA change type 6. Note these visual inspections are addressed by DOC 3.6.1.1-L.1, but only the frequency relaxation is mentioned. Also note that the ITS Section drop down list does not match BWR/4 Section 3.6 numbering Reply to response: An LA designation is appropriate because prescriptive details being removed are will be maintained per ILRT program requirements and associated commitments. Closeout remarks. Although DOC L.1 addresses the relaxation of the Frequency of the inspections, and the details of the inspection are contained in regulatory guidance referenced by the ILRT program, DOC L.1 could indicate that the inspection location information is redundant to ILRT program requiremnts and may therefore be deleted. How changes are categorized has varied during the history of reviewing ITS amendment applications, but is not usually a determining factor for accepting a change; the technical safety basis, if satisfactory, makes a change acceptable. In this case, the reduction in Frequency is acceptable for the general reasons that will be described for inspection frequency relaxations in the ITS amendment SE. The details for perfoming the inspections may be deleted because they are already a part of the ILRT program requirements, by reference. Therefore, the change addressed by DOC L.1 is acceptable, and this comment is closed.
Issue Date	10/17/2005

Close Date	02/24/2006 Resolution requires change to: None Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 01/23/2006	This response is provided to address the "Reply to response" portion of the NRC comment. The "Reply to response" NRC comment was added after Monticello responded to the original NRC comment. The "Reply to response" portion of the

	NRC comment is the NRC response to the original Monticello response. An "LA" type Discussion of Change (DOC) is used to only relocate information from the Current Technical Specifications (CTS) to another regulatory-controlled document, such as the USAR. It is not used when an actual change is being made to the relocated requirement. Monticello classified DOC L.1 (Attachment 1, Volume 11, Rev. 0, Pages 13 and 14 of 431) as an "L" type DOC because a technical change was being made to the CTS requirements; specifically, the Frequency of the visual inspections are being reduced. Therefore, the change cannot be classified as an "LA" type DOC; it must remain as an "L" type DOC. In addition, the two Monticello CTS requirements in question, CTS 4.7.A.1.c (Page 9 of 431) and 4.7.A.2.d (Page 7 of 431), require visual inspections of the "suppression chamber" and the "drywell." Both of these structures are part of the primary containment, thus Improved Technical Specification (ITS) SR 3.6.1.1.1, which requires visual inspections as part of the Primary Containment Leakage Rate Testing Program covers these two CTS requirements. Furthermore, this change is classified consistent with the most recently approved BWR ITS conversion, FitzPatrick Nuclear Power Plant. Prior to the conversion to the ITS at the Fitzpatrick Nuclear Power Plant, the Fitzpatrick CTS requirements associated with the primary containment visual inspections were similar to the Monticello CTS requirements. FitzPatrick CTS 4.7.A.1 included a requirement to perform a visual inspection of the accessible interior surfaces of the drywell and above the water line of the torus (suppression chamber) once per 24 months for evidence of deterioration. This FitzPatrick CTS requirement is consistent with Monticello CTS 4.7.A.1.c and CTS 4.7.A.2.d. In addition, FitzPatrick CTS 4.7.A.2.a required the performance of a visual examination and leakage rate testing of the Primary Containment in accordance with the Primary Containment Leakage Rate Testing Program. This requirement at FitzPatrick is consistent with Monticello CTS 4.7.A.2.a at Monticello. During the ITS conversion and as approved by the NRC in the Safety Evaluation for the ITS License Amendment (License Amendment 274, dated 7/3/02), an "L" type Discussion of Change was used to delete the 24 month drywell and suppression chamber visual inspections required by FitzPatrick CTS 4.7.A.1 (as it was being changed to the same Frequency that is required by 10 CFR 50, Appendix B, Option J (essentially three tests every 10 years)). This is the same type of change that Monticello is proposing in DOC L.1. Since the requirements for the containment visual examinations in the CTS (i.e., prior to ITS implementation) at FitzPatrick and Monticello are similar and since a "L" type change was approved by the USNRC for FitzPatrick, Monticello considers the ITS 3.6.1.1 DOC L.1 to be acceptable for this reason also (in addition to the reason provided in the first paragraph).
Licensee Response by Jerry Jones on 10/21/2005	Current Technical Specification (CTS) 4.7.A.1.c (Attachment 1, Volume 11, Rev. 0, Page 9 of 431) requires a visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made. CTS 4.7.A.2.a (Page 5 of 431) requires performance of required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program. CTS 4.7.A.2.d (Page 7 of 431) requires a visual inspection of the interior surfaces of the drywell for evidence of deterioration. All of the CTS requirements referenced above are identified to be incorporated into Improved Technical Specification (ITS) SR 3.6.1.1.1 (Page 16 of 431). ITS SR 3.6.1.1.1 states "Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program." While the specific details of the areas required to be visually inspected do not appear in ITS SR 3.6.1.1.1, the

	Surveillance Requirement specifically states to perform "required visual inspections." ITS 5.5.11.a (Attachment 1, Volume 17, Rev. 0, Page 92 of 143) specifies the Primary Containment Leakage Rate Testing Program requirements, and states that the program shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." Regulatory Position C.3 of RG 1.163 states that "Section 9.2.1, "Pretest Inspection and Test Methodology," of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." Furthermore, Nuclear Energy Institute (NEI) 94-01, Section 9.2.1 states "Prior to initiating a Type A test, a visual examination shall be conducted of accessible interior and exterior surfaces of the containment system for structural problems which may affect either the containment structure leakage integrity or the performance of the Type A test. This inspection should be a general visual inspection of accessible interior and exterior surfaces of the primary containment and components." Thus, the requirements of CTS 4.7.A.1.c and CTS 4.7.A.2.d are covered by ITS SR 3.6.1.1.1, via ITS 5.5.11.a and RG 1.163. The CTS 4.7.A.1.c requirement to visually inspect the "suppression chamber interior including water line regions and the interior painted surfaces above the water line" and the CTS 4.7.A.2.d requirement to visually inspect the "interior surfaces of the drywell" is covered by the Regulatory Guide phrase to visually inspect "accessible interior and exterior surfaces of the containment system." The phrase in CTS 4.7.A.2.d to visually inspect "for evidence of deterioration" is analogous to the Regulatory 1.163 phrase "for structural problems." Therefore, an "R" or "LA" Discussion of Change is not necessary.

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Close

NRC ITS TRACKING

NRC Reviewer

ID	**200510181436** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.4** ITS Number: **3.4.9** DOC Number: **None** JFD Number: **None** Bases JFD Number: **None** Page Number(s): **197**
Comment	**10 CFR 50.36(a) states that "[a] summary statement of the bases or reasons for ... [technical] specifications ... shall also be included in the application, but shall not become part of the technical specifications. Proposed Insert 1 for Bases Background 3.4.9 says in part "During ... hydrostatic testing, the reactor vessel shell temperatures ... shall be at or above the temperatures shown on the two curves of Figure 3.4.9-2 .... During heatup ... the RCS temperatures .... shall be at or above .... temperatures of Figure 3.4.9-3 ...." Inserts 1 and 2 for Bases 3.4.9 use the word "shall" in several other sentences and could be interpreted as stating requirements rather than providing reasons for a TS in accordance with 10 CFR 50.36(a). Monticello might consider revising the language in the Inserts 1 and 2 to make the inserts more informational rather than instructional. Alternately, the inserts from the bases could be moved into SR 3.4.9.x or TS Figure 3.4.9-x.**
Issue Date	**10/18/2005**

Close Date	**10/27/2005** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 10/20/2005	Improved Technical Specification (ITS) SR 3.4.9.1 (Attachment 1, Volume 9, Rev. 0, Page 185 of 255), in part, requires the verification that Reactor Coolant System (RCS) pressure and RCS temperature are within the "applicable limits" specified in Figures 3.4.9-2 and 3.4.9-3 and Note 2 to the SR states that Figures 3.4.9-2 and 3.4.9-3 shall be adjusted as required by Figure 3.4.9-1 for the reactor vessel shell and fluid temperature limits. ITS SR 3.4.9.2 (Page 185 of 255) requires the verification of RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.9-4 and the Note to the SR states that Figure 3.4.9-4 shall be adjusted as required by Figure 3.4.9-1 for the reactor vessel shell fluid temperature limits. ITS Figure 3.4.9-1 (Page 189 of 255) is a plot of Core Beltline Operating Limits Curve Adjustment versus Fluence. ITS Figure 3.4.9-2 (Page 190 of 255) is a plot of RCS Pressure versus Temperature Limits during Inservice Leak and Hydrostatic Testing. ITS Figure 3.4.9-3 (Page 191 of 255) is a curve of RCS Pressure versus Temperature Limits during Non-

	Nuclear Heatup and Cooldown. ITS Figure 3.4.9-4 (Page 192 of 255) is a plot of RCS Pressure versus Temperature Limits during Critical Operation. ITS 3.4.9 Bases Insert 1 (Page 197 of 255) states "During inservice leak and hydrostatic testing, the reactor vessel shell temperatures (reactor vessel shell adjacent to shell flange, reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region) shall be at or above the temperatures shown on the two curves of Figure 3.4.9-2, where the dashed curve, "RPV Core Beltline (Full Power Years)," is increased by the core beltline temperature adjustment from Figure 3.4.9-1. The reactor vessel bottom head temperature shall be at or above the temperatures shown on the solid curve of Figure 3.4.9-2, "RPV Remote from Core Beltline," with no adjustment from Figure 3.4.9-1. During heatup by non-nuclear means and cooldown following nuclear shutdown the RCS temperatures (reactor vessel shell adjacent to shell flange, reactor vessel bottom drain, recirculation loops A and B, reactor vessel bottom head) shall be at or above the higher of the temperatures of Figure 3.4.9-3 where the dashed curve, "RPV Core Beltline (Zero Full Power Years)," is increased by the expected shift in RTNDT from Figure 3.4.9-1." ITS 3.4.9 Bases Insert 2 (Page 197 of 255) states "During all operation with a critical reactor, the RCS temperatures (reactor vessel shell adjacent to shell flange, reactor vessel bottom head, and reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region) shall be at or above the higher of the temperatures of Figure 3.4.9-4 where the dashed curve, "RPV Core Beltline (Zero Full Power Years)," is increased by the expected shift in RTNDT from Figure 3.4.9-1." The Bases uses the word "shall" in several places. The reason the Bases is using this word is to match the requirements in ITS SR 3.4.9.1 and SR 3.4.9.2, including the Notes. These Bases inserts are describing what the actual SRs require; they are not adding any new requirements. The SRs require the limits of the Figures to be met, thus the use of the word "shall" is acceptable. Furthermore, the statements describing the acceptable regions of the Figures is only clarifying that the acceptable regions are above and to the left of the curves, which is readily obvious from the description of the LCO requirements and the SRs (i.e., the LCO is concerned with temperature being too low). Therefore, use of the word "shall" in the two Bases Inserts is not inconsistent with the 10 CFR 50.36(a) statement shown in the NRC reviewer's question.

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NRC ITS TRACKING

NRC Reviewer

ID	**200510251808** Conference Call Requested? **No**
Category	**Minor Technical**
ITS Information	ITS Section: **3.6** DOC Number: **A.1** JFD Number: **1** Page Number(s): **17** ITS Number: **3.6.1.1** Bases JFD Number: **None**
Comment	**TS 4.7.A.4.a.(2) states "Verify drywell to suppression chamber 'bypass leakage is less than that equivalent to one inch diameter orifice.'" What does this mean in terms used by STS SR 3.6.1.1.2, and why not present the criterion consistent with the STS, or at least put that information in the Bases?**
Issue Date	**10/25/2005**

Close Date	**11/16/2005** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 11/03/2005	Current Technical Specification (CTS) 4.7.A.4.a.(2) (Attachment 1, Volume 11, Rev. 0, Page 10 of 431) requires a demonstration that the "drywell to suppression chamber leakage is less than that equivalent to a one-inch diameter orifice." Improved Standard Technical Specification (ISTS) SR 3.6.1.1.2 (Page 17 of 431) requires verification that the "drywell to suppression chamber differential pressure does not decrease at a rate greater than [0.25] inch water gauge per minute tested over a [10] minute period at an initial differential pressure of [1] psid." ITS SR 3.6.1.1.2 has been modified to be consistent with CTS 4.7.A.4.a.(2) and therefore requires verification that "drywell to suppression chamber bypass leakage is less than that equivalent to a one inch diameter orifice." This change to ISTS SR 3.6.1.1.2 has been justified with Justification for Deviations (JFD) 1 (Page 18 of 431), which states "The brackets have been removed and the proper plant specific information/value has been provided." Monticello performs CTS 4.7.A.4.a.(2) by establishing a differential pressure of approximately 1 psid between the drywell and suppression chamber and then monitors the differential pressure decay for 25 minutes. The differential pressure decay data is plotted on a graph of differential pressure decay established for an equivalent one-inch diameter orifice. If the plotted data falls within the acceptable region of the curve, the leakage between the drywell and suppression chamber is less than that equivalent a one inch diameter orifice. The graph of differential pressure decay between drywell and suppression chamber for an equivalent one-inch diameter orifice is established so that the test can be performed at an initial condition from .75 psid to 1.3 psid. Thus, the Monticello CTS Surveillance Requirement is

	more flexible than the ISTS requirement, since the ISTS requirement is specific on the initial differential pressure condition that must be established. In addition, the ITS SR 3.6.1.1.2 Bases (Page 23 of 431) describes how the test is being performed, and states that a differential pressure must be established and that the test must be performed during a 25 minute period. Thus, information related to how the test is performed is provided in the ITS Bases. Therefore, Monticello desires to maintain control over the manner in which the Surveillance is performed in lieu of including the test parameters in the ITS. ITS SR 3.6.1.1.2 continues to include the acceptance criteria, consistent with the CTS requirement.

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NRC ITS TRACKING

NRC Reviewer

ID	200510281131 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 5.0 ITS Number: BSI 7 DOC Number: L.4 JFD Number: None Bases JFD Number: None Page Number(s): 101
Comment	This is a Beyond Scope Issue (TAC No. MC7610) You propose to delete CGCS from ITS 5.5.2, based upon the following: The Combustible Gas Control System is no longer required and has been removed from the Monticello Technical Specifications, and that this has been documented in License Amendment No. 138, dated May 21, 2004 (ADAMS Accession No. ML041180612). A plant modification is complete which removes all CGCS communication with primary containment. A plant modification is complete which removes Residual Heat Removal (RHR) System supply lines to the CGCS. The RHR system is included in ITS 5.5.2, but removing the RHR system from the CGCS removes any potential for highly radioactive fluid from entering CGCS via RHR. 1. In the application, it states that Technical Specification requirements governing the OPERABILITY of the Combustible Gas Control System (CGCS) have been previously removed. The Hydrogen Recombiner portion of the CGCS was removed from Technical Specifications; however, it appears much of the CGCS piping appears to still exist. The statement was also made that a plant modification was completed which removed all CGCS communication with primary containment. The Containment Atmosphere Monitoring System (CAMS) appears to still be available to place in service, and the CAMS appears to be part of the CGCS, in that the inlet and return lines tap off the 4" CGCS process line. Although the Hydrogen Recombiner portion of the CGCS may be retired in place, a portion of the CGCS still appears to communicate with primary containment and may still be subject to highly radioactive fluid, as in the case when the CAMS is placed in service post-accident. 1) Provide further clarification as to the status of the CGCS and the system current system interface with primary containment. 2) Provide justification as to how the CGCS may be completely removed from the proposed program requirements of ITS 5.5.2 if it may still be subject to potentially radioactive fluid. 2. If a proposed response to Question 1 is utilizing the ITS 5.5.2 requirement related to Process Sampling equipment monitoring, then 1) Is the CAMS being monitored as part of your CTS 6.8.B program? 2) Will it be monitored under the proposed ITS 5.5.2 3. You state that the RHR system cooling water supply lines have been eliminated to the CGCS. If the CGCS supply valves (CGC-1-1 or CGC-1-2) are open or can re-opened, then it appears that the RHR system could potentially communicate with the CGCS. 1) Explain how the RHR supply line is eliminated and where is this isolation (the supply

	isolation valve or at the recombiner skid)? **2) Are CGC-1-1 and CGC-1-2 administratively controlled closed and/or another method used (i.e. a blank flange installed) for positive isolation?** **4. What will be the process for maintaining the CGCS and associated RHR system configuration, such that any future modifications do not inadvertently impact this proposed change?**
Issue Date	**10/28/2005**

Close Date	**02/03/2006**	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 11/08/2005	Improved Technical Specification (ITS) 5.5 Discussion of Changes (DOC) L.4 (Attachment 1, Volume 17, Rev. 0, Page 76 of 143) states that a plant modification has been completed that removes all communication between the Combustible Gas Control System (CGCS) and the containment and eliminated the Residual Heat Removal (RHR) System cooling water supply lines to the CGCS. This plant modification was documented under Modification Procedure 03Q145, and the piping modifications were performed during refueling outage RF022. The CGCS inlet and return lines piping was cut and capped on the containment side of the Containment Atmosphere Monitoring System (CAMS) sample connection points. The RHR System cooling water supply lines were cut on the CGCS side of valves CGC-1-1 and CGC-1-2. Furthermore, a high point vent connection was added to the RHR System just downstream of CGC-1-1 and CGC-1-2 (and the valves were redesignated as RHR System valves) and two vent valves were added and the vent capped. In addition, during the development of the Monticello response to this question, it was noted that the Current Technical Specifications (CTS) Markup page associated with this change was not correctly annotated. The words "Combustible Gas Control," in CTS 6.8.B (Page 55 of 143) should have been deleted and annotated with DOC L.4. This will be corrected as shown in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	**200510281147** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.4** ITS Number: **BSI 6** DOC Number: **M.1** JFD Number: **None** Bases JFD Number: **None** Page Number(s): **5**
Comment	**This is a Beyond Scope Issue (TAC No. MC7609) 1. Describe how the operator uses the Bases, Technical Requirements Manual (TRM) and Technical Specifications (TS) to operate the reactor during a normal and anticipated operational occurrences with respect to this proposed TS change. As required in the regulations 10 CFR 50.36, TS should play the major role not Bases or TRM. Describe the technical basis for inserting control rods or raising the speed of an operating recirculation pump not being provided in LCO 3.4.1 Action A.1, but instead being described in the Bases, and APRM Rod Block setpoints are not provided in LCO 3.4.1.c but in the TRM. 2. Describe in detail the calculation procedures used to update the Stability Buffer and Exclusion Region of the power-to-flow map in the COLR since the Monticello?s first implementation of Long-Term stability Option ID solution. Describe any changes of the calculation method with respect to Part 21 DIVOM curve issue and any functional improvement of a stability monitoring system used in Monticello plant (such as SOLOMON or equivalent). 3. Describe in detail the relationship between the Stability Buffer Region and the power distribution control in the proposed SR 3.4.1.2, and the actions to be taken if operation is in the Stability Buffer Region of the power-to-flow map. Also, please identify that the power distribution control is a cycle-specific core operating parameter. If the answer is no, please justify why it is specified in COLR.**
Issue Date	**10/28/2005**

Close Date	**03/16/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/01/2005	NOTE: At the request of the NRC, this response was edited for clarity. 1. Improved Technical Specification (ITS) LCO 3.4.1 (Attachment 1, Volume 9, Rev. 0, Page 11 of 255) requires the plant to be operated in the Normal Region of the power to flow map specified in the COLR. Furthermore, the LCO 3.4.1 Note (Page 12 of 255) allows operation within the Stability Buffer Region of the power to flow map specified in the COLR provided the power distribution controls specified in the COLR are in effect. If this requirement is not met, ITS 3.4.1

	ACTION A (Page 12 of 255) requires actions to be initiated to restore operation to within the Normal Region of the power to flow map specified in the COLR. All of the above ITS requirements are consistent with the Current Technical Specifications. While the CTS also includes the specific manner in which to restore operation to within the Normal Region, it was not included in the ITS as justified in Discussion of Change (DOC) LA.1 (Page 9 of 255). The specific details were moved to the ITS Bases since it is common to not include procedural details for meeting Technical Specification requirements in the ITS. For example, the Improved Standard Technical Specifications (ISTS) 3.1.3, which provides requirements for Control Rod OPERABILITY, does not include the details for disarming control rod drives, which was previously in the Technical Specifications. The details for disarming control rod drives is now located in the ISTS Bases. Furthermore, recent ITS conversions for James A. FitzPatrick Nuclear Power Plant and LaSalle Units 1 and 2 moved similar information concerning how to exit restricted stability regions to the associated ITS Bases, and classified the movement using an LA type DOC. This is the reason Monticello did not consider the movement of this information to the ITS Bases a beyond scope issue. 2. The stability monitoring regions are currently described in the Monticello COLR, Section 6.0 (NAD-MN-010, Monticello Cycle 23 COLR, Rev. 1). This document has been previously provided to the NRC. The stability monitoring requirements were added to the Current Technical Specifications (CTS) in License Amendment 97, as approved by the NRC in the Safety Evaluation Report dated 9/17/96. The change described in DOC M.1, which is the beyond scope issue, does not change the stability monitoring regions; it only justifies the addition of a Surveillance Requirement to ensure plant operation is within the currently approved stability monitoring regions. 3. As stated above, proposed SR 3.4.1.2 only verifies that plant operation is in a currently approved stability monitoring region. If the Surveillance is not met, ITS 3.4.1 ACTION A is to be taken. ITS 3.4.1 ACTION A is consistent with the current requirements (CTS 3.5.F.1 and 3.5.F.2); no changes to these currently approved actions are being proposed, with the exception of moving certain information to the ITS Bases as described in DOC LA.1 and discussed above in the response to question 1. Furthermore, the CTS currently allows the power distribution controls to be defined in the COLR, as approved in License Amendment 97. Therefore, the Monticello ITS will retain these controls in the COLR.
Licensee Response by Jerry Jones on 03/07/2006	This issue was discussed at the weekly NRC/Monticello phone conference where updates on the ITS conversion are provided. The boundaries of the stability buffer and exclusion regions on the Monticello COLR power/flow map (COLR Figures 6 and 7) are calculated each cycle by Global Nuclear Fuels (GNF) when they perform our cycle-specific licensing calculations. GNF uses NRC-approved methods to perform these calculations. The COLR itself is then updated by NMC to reflect the results of the licensing calculations, using our standard procedures for making cycle-specific updates to the COLR. Any issues associated with DIVOM (a parameter/methodology within the process used to determine the boundaries of the stability regions) are addressed directly by GNF through their internal processes for dealing with the effects of various issues on their licensing methodologies. Although NMC is generally aware of ongoing DIVOM issues, we do not directly drive GNF?s licensed calculation methods. The Option I-D stability monitoring system originally used at Monticello was the GNF SOLOMON software. In April 2005, SOLOMON was replaced by SIMULATE-3K (S3K) when we changed our reactor core monitor from 3D-Monicore to Gardel. S3K is Studsvik/Scandpower software that performs a three-dimensional time domain calculation of reactor neutronic and thermal-hydraulic response to

	determine both global and regional stability decay ratios. These calculations are performed automatically approximately every three hours by Gardel (using S3K) to ensure that the decay ratios reflect current core conditions. The licensed operators can demand a new calculation of the decay ratios at any time. The results of the most recent decay ratio calculation are displayed on the Gardel live user interface. The licensed operators can then use the calculated decay ratios to evaluate whether reactor operation is within the allowed area of the Monticello stability decay ratio map (COLR Figure 7).

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NRC ITS TRACKING

NRC Reviewer

ID	**200510281240** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.3** DOC Number: **M.3** JFD Number: **None** Page Number(s): **683** ITS Number: **BSI 1h** Bases JFD Number: **None**
Comment	**This is a Beyond Scope Issue (TAC No. MC7604) 1. In the ITS, Table 3.3.8.1-1, the allowable value of Loss of Voltage Protection is between greater than or equal to 2345 V and less than or equal to 2905 V. This corresponds to an equivalent trip setting of 2625 plus or minus 280 V. In the Current Technical Specifications (CTS), the trip setting is 2625 plus or minus 175 V. Please provide a copy of the drift calculation based upon which the proposed allowable value for Loss of Voltage Protection (change in drift value) in the ITS has been proposed. 2. In the ITS, Table 3.3.8.1-1, the allowable value of Degraded Voltage Protection is between greater than or equal to 3909 V and less than or equal to 3921 V with a time delay of greater than or equal to 8.8 seconds and less than or equal to 9.2 seconds. This corresponds to an equivalent trip setting of 3915 plus or minus 6 V and a time delay of 9.0 plus or minus 0.2 seconds. In the CTS, the trip setting is 3915 plus or minus 18 V and a time delay of 9.0 plus or minus 1.0 second. Please provide a copy of the drift calculation based upon which the proposed allowable values for Degraded Voltage Protection (change in the drift values) in the ITS have been proposed.**
Issue Date	**10/28/2005**

Close Date	**02/02/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 11/15/2005	The NRC reviewer requested a copy of the drift calculation supporting the new Allowable Values for the Loss of Voltage and Degraded Voltage Functions. The NRC reviewer identified the Allowable Values for both of these Functions are being changed as part of the conversion to the Monticello Improved Technical Specifications (ITS). However, the Loss of Voltage Function Allowable Value is not being changed as part of the Monticello ITS conversion. The change to the Current Technical Specifications (CTS) Table 3.2.6 Loss of Voltage Protection Function (Attachment 1, Volume 8, Rev. 0, Page 683 of 760) identifies the change to the Allowable Value is justified in Discussion of Change (DOC) A.7. DOC A.7 (Page 690 of 760) states that the change in the Allowable Value is consistent with the Technical Specifications Change Request submitted to the

	NRC for approval in NMC letter L-MT-04-036, from Thomas J. Palmisano (NMC) to NRC, dated June 30, 2004. This change has subsequently been approved by the NRC as part of Monticello License Amendment 143, dated September 30, 2005. Therefore, since the Loss of Voltage Allowable Value change has already been approved by the NRC, it is not part of the Monticello ITS conversion and should not be considered a beyond scope issue. Thus, no drift calculations are being provided. The changes to the Degraded Voltage Allowable Values (Page 683 of 760) are justified by DOC M.3 (Page 691 of 760), and are being changed as part of the Monticello ITS conversion. The instrument drift analysis for both the Voltage Function and the Time Delay Function will be provided to the NRC reviewer. [added on 11/17/05] Based on a e-mail from the NRC to Monticello, the NRC reviewer requested the setpoint calculation in addition to the drift analysis. Therefore, the setpoint calculation for the Degraded Voltage Function (voltage and time delay) will be provided to the NRC reviewer. [added on 12/7/05] The setpoint calculation has been provided in the attachment to this response.
Licensee Response by Jerry Jones on 04/08/2006	As a result of modifications to Instrument Setpoint Calculation for the 4.16 KV Degraded Voltage channels, the OPERABILITY limit (Allowable Value) must be changed. The changes to the Degraded Voltage Allowable Values (Attachment 1, Volume 8, Rev. 0, Pages 693 and 697 of 760) are provided in the attachment to this response. The previously submitted Allowable Values of greater than or equal to 3909 V and less than or equal to 3921 V have been changed to greater than or equal to 3913 V and less than or equal to 3927 V. The major contributor to the changes is the result of a re-characterization of the primary element accuracy associated with the potential transformer. The revised calcuation (CA-92-220 Rev. 1) has also been provided in the attachment to this response.

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CALCULATION COVER SHEET

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Title *Instrument Setpoint Calculation 4.16KV Degraded Voltage* CA- 92 - 220 Rev. 1

10 CFR50.59 Screening or Evaluation No: *SCR-05-0242, Rev. 1*

Associated Reference(s):

Does this calculation:	YES	NO	Calc No(s), Rev(s), Add(s)
Supercede another calculation?	☐	☒	
Augment (credited by) another calculation?	☐	☒	
Derive inputs from another calculation?	☐	☒	
Affect the Fire Protection Program per Form 3765?	☐	☒	If Yes, attach Form 3765
Affect piping or supports?	☐	☒	If Yes, attach Form 3544
Affect IST Program Valve or Pump Reference Values, and/or Acceptance Criteria?	☐	☒	If Yes, inform IST Coordinator and provide copy of calculation

*This is a major rewrite, therefore, no sidebars are required.

List all documents/procedures that are based on this calculation (include revision):

0302, Rev. 19

List all plant procedures used to ensure inputs/assumptions/outputs are maintained (include revision):

MWI-3-M-2.01, Rev. 9

What Systems or components are affected?

System Code(s) (See Form 3805): 4KV

Component ID's (CHAMPS Equip): 127-5A, -5B, -5C, 127-6A, -6B, -6C

DBD Section (if any): T.17, B.09.06

Topic Code (See Form 3805): N/A

Structure Code (See Form 3805): N/A

Other Comments:

Approved (Signatures available in Master File)

1. PURPOSE

The purpose of this calculation is to assure that the current setpoints and Allowable Values are conservative or to establish new setpoints and Allowable Values for the Degraded Voltage Relays listed below. This calculation establishes the bases for these settings and provides tolerances to be used in calibration procedures.

- 127-5A
- 127-5B
- 127-5C
- 127-6A
- 127-6B
- 127-6C

The Potential Transformers (PTs) addressed by this study are BUS-15/POT and BUS-16/POT.

Revision 1 of this calculation is produced to support the Improved Technical Specifications project. This calculation derives the necessary Allowable Values and associated setpoints.

2. METHODOLOGY

This calculation is performed in accordance with General Electric Setpoint Methodology (ESM-03.02-APP-I -- Input 4.1) and and the project Drift Analysis Methodology (ESM-03.02-APP-III – Input 4.13).

The General Electric Setpoint Methodology is a statistically based methodology. It recognizes that most of the uncertainties that affect instrument performance are subject to random behavior, and utilize statistical (probability) estimates of the various uncertainties to achieve conservative, but reasonable, predictions of instrument channel uncertainties. The objective of the statistical approach to setpoint calculations is to achieve a workable compromise between the need to ensure instrument trips when appropriate, and the need to avoid spurious trips that may unnecessarily challenge safety systems or disrupt plant operation.

The project Drift Analysis Methodology prescribes how actual As Found/As Left data is used to characterize instrument performance such that instrument and loop accuracy performance can be based on actual field observations to provide the most realistic modeling of instrument performance.

Drift values for the relays covered by this calculation were determined in Attachments 1 and 3 (voltage function) and Attachments 2 and 4 (time delay). The determination of relay drift values used in this calculation is performed in accordance with ESM-03.02-APP-III (Input 4.13). Since calibration intervals are not changing for the relays covered by this calculation, a time dependency analysis is not required.

<u>Voltage Function</u>
Note that Section 6.2.1 of Input 4.13 (drift analysis ESM-03.02-APP-III) states that "Only the Vendor Drift and Drift Temperature Effect terms may be replaced with the analyzed drift value for the Technical Specifications calculations performed per the GE setpoint methodology". For the voltage function for this calculation, the Vendor Accuracy term was computed using 2 methods:

(1) Using the vendor accuracy components from Input 4.8 (vendor technical manual) per the restriction imposed on Technical Specification calculations by Input 4.13 described above.

The vendor technical manual (Input 4.8) for these relays provides 3 different components for the VA term. Based on the calibration method of Input 4.4, only 2 of the terms are applicable to this calculation – the pickup and dropout setting repeatability at constant temperature and constant voltage and the pickup and dropout repeatability over dc power range of 100-140 volts. The pickup and dropout settings with respect to printed dial markings is not applicable because the setting is based on the measured value rather than the printed dial markings. Conservatively combining the two applicable terms using the square root of the sum of the squares yields a VA result of ±0.424 Vac (refer to Section 7.2.1.3).

(2) Using the broader guidance, also from Section 6.2.1 of Input 4.13, that allows the Analyzed Drift (AD) to characterize not only Vendor Drift (VD) but also Vendor Accuracy (VA) and M&TE (or calibration error).

Using actual As Found/As Left data for the installed relays yields a random accuracy term of ±0.0552 Vac with a negligible bias.

Thus, the vendor specified accuracy exceeded the observed installed accuracy by a factor of approximately 8 to 1.

Although method (1) is specifically prescribed by Input 4.13 for Technical Specification calculations, for this calculation method (2) is justified because:
- Method (1) results in an unrealistic and substantially greater error than has actually been observed; and
- Method (2) provides a Vendor Accuracy (VA) term based on actual observed behavior that is conservatively considered to be a more accurate characterization of relay performance.

Time Delay Function
The method used to determine the time delay Vendor Accuracy (VA) is consistent with the methodology of 6.2.1 of Input 4.13 (drift analysis ESM-03.02-APP-III) for Technical Specifications calculations. Vendor Accuracy is derived from vendor specifications.

The methodology for determining instrument setpoints is not described in the USAR or its references.

3. ACCEPTANCE CRITERIA

The setpoint and instrument settings must provide assurance that the Analytical Limit will not be exceeded when all applicable instrumentation uncertainties are considered.

4. INPUTS

4.1 Engineering Standards Manual ESM-03.02-APP-I, Appendix I (GE Methodology Instrumentation & Controls), Revision 3. The ESM provides plant specific guidance on the implementation of the General Electric guidelines (Reference 5.1) and methodology (Reference 5.2).

4.2 Monticello Technical Specifications, Amendment 143.

Section	Setting	Function
Table 3.2.6 Item 1	3915 ±18 Volts 9 ± 1 Seconds	Safeguards Bus Degraded Voltage Protection
Bases Section 3.2 Deviation Table	≥ 3897 Volts (trip) ≤ 3975 Volts (reset) 5 Sec. ≤ Time delay and 10 Sec. ≥ Time delay	Instrumentation for Safeguards Bus Protection – Degraded Voltage
Table 4.2.1 Safeguards Bus Voltage, Item 1	Quarterly	Calibration

Per the surveillance procedures (Input 4.4), calibration of the Degraded Voltage relays fulfills the requirements of Technical Specification Table 4.2.1, Item 1, for calibration, and is to be performed Quarterly.

4.3 Monticello Component Master List (CML). The CML contains instrument information relating to the installed equipment as listed in Sections 7.2.1.1 and 7.6.2.1.

4.4 0302, Revision 18, Safeguard Bus Degraded Voltage Protection-Relay Unit Calibration. This procedure provides the calibration requirements for the subject degraded voltage relays. The procedure identifies the calibrating instrument as a Datron Model 1271 Digital Multimeter, and gives the following information.

Voltage Function	
Nominal Setpoint (127-5A, 5B, 5C, 6A, 6B, 6C)	111.96 Volts
As Found Values	111.34 ≤ DO ≤ 112.37 Volts 3897 ≤ DO ≤ 3933 Bus Voltage
As Left Range	111.96 ± 0.05 Volts or 111.91 ≤ DO ≤ 112.01 Volts
Time Delay	
Nominal Setpoint (127-5A, 5B, 5C, 6A, 6B, 6C)	9.0 Seconds
As Found Range	8.0 ≤ time delay ≤ 10.0 Sec.
As Left Range	9.0 ± 0.10 Seconds or 8.9 ≤ time delay ≤ 9.1 Sec.

4.5 NX-9532-1, Revision 0, "600 V Through 15 KV Butyl-Molded & Compound Filled Transformers," GEH-230AA, "Instructions - Instrument Transformers – Butyl-Molded and Compound-Filled, 600-V Through 15-KV," Dated May, 1968. This manual shows that the Potential Transformers, which supply the input signal to the Loss of Voltage relays, were produced in accordance with the American Standards for Instrument Transformers, ASA C57.13.

4.6 American National Standard ANSI/IEEE C57.13-1978, "Requirements for Instrument Transformers." Per Section 5 of this standard, "Accuracy Classes for Metering Service", Table 6 lists the metering accuracy classes as 0.3%, 0.6% and 1.2%.

4.7 NF-36397, Revision Y (Passport Revision 075), "Monticello Nuclear Generating Plant Schematic – Meter & Relay Diagram – 4160V System – Buses #11, #12, #13, #14, #15, #16." This diagram shows that the potential transformers used for buses 15 and 16 for the Degraded Voltage relays is 4200-120V, or has a winding ratio of 35:1.

4.8 NX-16951, Revision 0, "Single Phase Undervoltage Relays, Brown Boveri, Inc.," ITE-27N Undervoltage Relay. This vendor technical manual provides the following accuracy and burden terms.

For the voltage function, these accuracy terms are all given as %. For the purposes of this calculation the % is set as % of span.

Accuracy Term	Value
Pickup and dropout settings, repeatability at constant temperature and constant control voltage	±0.2% of 150 Volt Span = 0.3 Volts
Pickup and dropout settings, repeatability over dc power range of 100-140 volts	±0.2% of 150 Volt Span = 0.3 Volts
Pickup and dropout settings, repeatability over temperature range of 32°F to 104°F	±0.2% of 150 Volt Span = 0.3 Volts over 72°F or 0.00417 Volts/°F

For the time delay function, the accuracy terms for the definite time relays is specified as ± 20 milliseconds or ± 10%, whichever is greater.

The ± 10% is based on the use of the printed dial settings during the calibration, while ± 20 milliseconds is based on using the measured value. Since the measured value is used, per Input 4.4, the value for VA is:

VA = ± 20 milliseconds

There is not an accuracy temperature effect specified for the time delay function.

Burden is less than 1 VA at 120 VAC.

4.9 Monticello USAR Section 8.3, "Auxiliary Power System," and Section 8.4, Revision 22, "Plant Standby Diesel Generator Systems." These sections of the USAR provide description of the functions involved with Degraded Voltage protection and information related to the bases for the settings.

4.10 Letter from R. C. Anderson of Bechtel Power Corporation to D. Antony of Northern States Power Company, Subject: Job 10040, Monticello Nuclear Generating Plant – Unit 1, Northern States Power Company - Equipment Performance Under Degraded Grid Voltage Conditions, Dated October 7, 1976.

This document provides the majority of the bases for the logic in establishment of the Degraded Voltage relay settings.

4.11 MWI-3-M-2.01, Revision 8, "AC Electrical Load Study."

Per Appendix II, Note 2 of the AC Electrical Load Study, the Degraded Voltage limits are established as follows:

The upper Degraded Voltage limit for motor starting studies is the Technical Specification Setpoint plus the high side setting tolerance:

3915 + 18 = 3933 Vac (Bus Voltage);
Converting to relay voltage yields:
3933/35 = 112.3714 Vac

The lower Degraded Voltage limit for motor starting studies is the Technical Specification Setpoint minus the low side setting tolerance:

3915 - 18 = 3897 Vac (Bus Voltage);
Converting to relay voltage yields:
3897/35 = 111.3429 Vac

These limits are established as the Lower and Upper Analytical Limits for the Degraded Voltage Function.

This Input will require revision since the above limits will not be included in the Improved Technical Specifications. Future Need 9.4 has been included to update the MWI.

4.12 Passport Equipment/Component Header. This database contains the manufacturers and model numbers of the Degraded Voltage relays analyzed in this calculation.

4.13 Engineering Standards Manual ESM-03.02-APP-III, Appendix III (Drift Analysis (Instrumentation and Controls), Revision 4. Section 6.2.1 of this ESM states that the Analyzed Drift term may be incorporated into the calculation, setting the Vendor Accuracy, M&TE (or calibration error), and the drift terms for the analyzed devices to zero.

4.14 MPS-0538, Revision 14, Bechtel Specification M-118, Heating, Ventilating, and Air Conditioning Systems. Given the values shown below, this calculation conservatively uses an ambient temperature range of 60°F to 104°F for the development of instrument uncertainties for the Degraded Voltage Relays.

Temperature	
Winter	60°F
Summer	104°F

4.15 Letter from D. Musolf, NSP, To Director NRR, "Reanalysis of Adequacy of Station Electric Distribution System Voltages," dated December 30, 1983

As described in Section 3.2 of Input 4.15, the plant electrical distribution system was modeled using a computer application that used the Newton-Raphson iterative program for load analysis. Various cases were then run using the calculated load distribution for the LOCA, maximum and minimum auxiliary loads. The results of this effort established 3897 Vac (bus voltage) as the voltage necessary on the essential safeguards buses 15 and 16 to maintain the minimum allowable voltage on the 120 Volt instrument buses with:

1. Full station auxiliary load
2. ECCS actuation
3. Load shed per plant design

Using the assumed ±18 Vac (bus voltage) setting tolerance, a relay setpoint of 3915 Vac (3897 + 18 Vac) was derived. To assure relay reset in subsequent analysis, the 18 Vac positive side tolerance and the 42 Vac reset band were added on, resulting in a 4 KV bus voltage of 3975 Vac or above. Thus any transient case which results in a voltage recovery to 3975 Vac or above in less than 10 seconds will assure that the Degraded Voltage protective scheme will not be actuated.

4.16 Technical Manual NX-17343, Rev 0, "Wavetek Datron Model 1271 Digital Multimeter." This manual provides the specifications for the Measurement and Test Equipment used for the calibration of the subject relays.

4.17 Technical Manual NX-9064-90-4, Rev. 11, "Relays-General Electric (Volume 4)." This manual provides the burden data for the GE NGV15A undervoltage relays.

Burden is 4.2 watts (unity power factor) at 120 volts.

4.18 Technical Manual NX-9064-90-6, Rev. 0, "Protective Relays - Non-General Electric." This manual provides the burden data for the ITE-27H undervoltage relays.

Burden is 1.2 VA, 1.0 power factor, at 120 volts.

4.19 Technical Manual NX-9308-1, Rev. 1, "Tranzducers." This manual provides the burden data for the GE voltage transducers.

Burden is 5.0 VA.

4.20 Yokogawa Switchboard Instruments Specifications. The installed panel voltmeters and synchroscope were originally supplied by General Electric. This instrument line has since been purchased by Yokogawa. The specifications in the Yokogawa documentation are applicable to the installed equipment. The applicable page from Yokogawa is included as Attachment 10.

AB-40 Voltmeter burden is 0.51 VA
Syncroscope (Incoming) burden is 5.6 VA

4.21 Excerpt from Doble F2000 Operating Manual, included as Attachment 7. This manual specifies an accuracy of the digital timer included with the Doble F2000 as ± 0.01% of Reading and ± 1 Least Significant Digit (L.S.D.). The range of the timer is 0-999.99 seconds for a 9 second measurement.

4.22 GE Meter, Instrument's Transformers Buyer's Guide. This manual includes information on GE transformers that is used in the determination of the potential transformer's primary element accuracy. Selected pages from this document are included in Attachment 8.

4.23 Characteristic Fan Curve, Type JVM-3 & JVW-3 Potential Transformer. The characteristic fan curve used in the determination of the potential transformer's primary element accuracy. This document is included in Attachment 8.

4.24 Work Order 0401473 (Passport W/O 00135888), Obtain BUS-15/POT Nameplate Information. The Bus-15 potential transformers were determined to be Type JVM-3, Catalog Number 643X94.

5. REFERENCES

5.1 GE-NE-901-021-0492, DRF A00-01932-1, Setpoint Calculation Guidelines for the Monticello Nuclear Generating Plant, October 1992.

5.2 General Electric Instrument Setpoint Methodology, NEDC-31336P-A, September 1996.

5.3 Generic Letter 91-04, Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.

5.4 Condition Report 02001013 (AR00628275), Documentation of NRC Resident Question Regarding the Application of Tech Spec Deviations in As-Found Acceptance Criteria.

5.5 DBD T-17, Design Basis Document for Electrical Design Considerations, Revision D (Passport Revision 75).

5.6 Calibration Certificate for XDV-1271, Serial No. 025811-8, Dated 4-30-2004, Seq P050602065, Loc. No. M05807-2436.

6. ASSUMPTIONS

6.1 Per Work Order 0401473 (Input 4.24), the Bus 15 transformer is a Type JVM-3 Potential Transformer. At the time of the preparation of this calculation, the Bus 16 transformer is not available for inspection. It is assumed that the Bus 16 transformer is identical to the Bus 15 transformer for the purposes of this analysis. Future Need 9.3 has been identified to obtain the nameplate information from the Bus-16 transformer.

6.2 From Input 4.7, there are two potential transformers used in the metering circuits with the loads split between the two transformers. For this calculation, the total potential transformer burden is assumed to apply to each potential transformer. This is conservative since it maximizes the considered burden.

6.3 The calibration test meter is not calibrated in-house but is shipped offsite for calibration. Reference 5.6 is an example calibration certificate. The certificate makes the following statement, "All services provided a 4:1 uncertainty ratio (or greater) unless otherwise stated." Due to the varying range of uncertainties of the instrument, based on range, it is difficult to ensure with 100% confidence that a full 4:1 uncertainty ratio is maintained. Therefore, a conservative ratio of 2:1 will be considered to determine the calibration standard error used in Section 7.2.1.6.

7. ANALYSIS

7.1 Instrument Channel Arrangement

<u>Channel Diagram</u>: [4.16 KV Bus 15 or 16 Potential Transformer] → [Degraded Voltage Relays] → [Degraded Voltage Trip]

<u>Definition of Channel:</u> The Potential Transformer, with a 35:1 winding ratio, produces an approximate 120 Vac signal from a 4.2 KV voltage on the 4.16 KV

bus. This voltage is sensed by the Degraded Voltage relays, which provide the Degraded Voltage trip.

Functional Description: The Degraded Voltage relays monitor and detect the degraded voltage condition on the offsite power system and initiate the necessary actions required to transfer the essential buses #15 and #16 to the onsite system. The following description is derived from Input 4.10 and sub-sections of Reference 5.5.

Starting of the EDGs is initiated by a degradation or loss of voltage on an essential 4160 Vac bus. Automatic starting is also initiated by low-low reactor water level or high drywell pressure.

Although an automatic start of the EDGs has been initiated, there may have been no loss of voltage on the safety related 4 KV buses, or an automatic transfer to another source may have been effected, in which case the running generators are held in reserve during the emergency period. Manual control is then employed for additional load switching.

Transfer of the essential buses to either of the emergency power sources, the reserve auxiliary transformer (1AR) or the EDGs, will occur due to loss of essential bus voltage or degraded voltage conditions on the essential bus. Transfer of the essential buses to the 1AR transformer will normally occur on loss of voltage or degraded voltage conditions. If the 1AR no-load voltage is unacceptable, or if the essential buses are being supplied from the 1AR transformer when the loss of voltage or degraded voltage condition occurs, a transfer to the EDGs will take place.

If the essential buses are still de-energized when the diesels have accelerated, automatic relaying will remove unnecessary loads and disconnect the essential buses from the normal auxiliary system prior to energizing the essential buses from the EDGs. If a loss of coolant accident condition is indicated, Core Spray and RHR Systems are started. These pumps are started in sequence in order to prevent stalling of the diesel engine.

7.2 Instrument Definition and Determination of Device Error Terms – Voltage Function

7.2.1 DEVICE 1

7.2.1.1 Instrument Definition:

		Reference
Component ID:	**BUS-15/POT**	
Location:	Turbine Building, Elevation 911', Lower Level, Lower 4KV Room	4.3
Manufacturer:	General Electric	4.5
Model Number:	Type JVM-3, Cat. 643X94	4.5, 4.24
Ratio:	35:1	4.7
Input Signal:	4200 Vac - Nominal	4.7
Output Signal:	120 Vac - Nominal	4.7

		Reference
Component ID:	**BUS-16/POT**	
Location:	Turbine Building, Elevation 931', Ground Floor, Upper 4KV Area	4.3
Manufacturer:	General Electric	4.5
Model Number:	No Specific – Manual GEH-230AA	4.5, 6.1
Ratio:	35:1	4.7
Input Signal:	4200 Vac - Nominal	4.7
Output Signal:	120 Vac - Nominal	4.7

		Reference
Component ID:	**127-5A, -5B, -5C**	
Location:	Turbine Building, Elevation 911', Lower Level, Lower 4KV Room, Cubicle 152-510	4.3
Manufacturer:	ITE	4.12
Model Number:	27N211T4175	4.12
Setpoint:	3918.6 Volts AC 3918.6/35:1 = 111.96 Volts AC	4.4
Output Signal:	Contact Output	4.4

		Reference
Component ID:	**127-6A, -6B, -6C**	
Location:	Turbine Building, Elevation 931', Lower Level, Ground Floor 4KV Room, Cubicle 152-601	4.3
Manufacturer:	ITE	4.12
Model Number:	27N211T4175	4.12
Setpoint:	3918.6 Volts AC 3918.6/35:1 = 111.96 Volts AC	4.4
Output Signal:	Contact Output	4.4

7.2.1.2 Process and Physical Interfaces:

Calibration Conditions:		Reference
Temperature:	65 to 90°F	5.1
Current Surveillance Interval for Loss of Voltage Relays:	Quarterly	4.2
Proposed Surveillance Interval for Degraded Voltage Relays:	Quarterly	Note: The Degraded Voltage Relay calibration interval is not being extended based on this calculation.

Normal / Trip Plant Environmental Conditions:		Reference
Temperature Range:	60°F to 104°F	4.14

Seismic Conditions:		Reference
OBE Prior to Function:	N/A	N/A
OBE During Function:	N/A	N/A

These relays respond to a degraded voltage condition that is not related to any DBA or seismic event. Therefore, seismic conditions are not required to be determined for the Degraded Voltage relays.

Process Conditions:		Reference
During Calibration:	N/A	N/A
Worst Case:	N/A	N/A
During Function:	N/A	N/A

During the event when these devices are required, the Degraded Voltage relays are not subjected to process conditions (static pressure, overpressure, elevated temperatures, etc.) that would affect the accuracy of the instrument.

7.2.1.3 Individual Device Accuracy:

Term	Value	Sigma	Reference
VA:	±0.0552 Vac	2	Attachment 1, Section A1.8; Note 1
ATE:	±0.1835 Vac	2	Note 2
OPE:	NA	NA	Note 3
SPE:	NA	NA	Note 7
SE:	0	NA	Note 6
RE:	0	NA	Note 6
HE:	0	NA	Note 6
PSE:	NA	NA	Note 4
REE:	NA	NA	Note 5

Note 1: Per Input 4.8, there are two vendor accuracy components encompassed by VA:

- Pickup and dropout settings at constant temperature and constant control voltage.
- Pickup and dropout settings over the dc power range of 100-140 volts.

These two terms, and their associated values are:

Accuracy Term	Value	Designated
Pickup and dropout settings, repeatability at constant temperature and constant control voltage	±0.2% of 150 Vac Span = 0.3 Vac	VA_1
Pickup and dropout settings, repeatability over dc power range of 100-140 volts	±0.2% of 150 Vac Span = 0.3 Vac	VA_2

$$VA = \pm\sqrt{VA_1^2 + VA_2^2}$$

$$VA = \pm\sqrt{0.3^2 + 0.3^2}$$

$$VA = \pm 0.424 \text{ Vac}$$

The Analyzed Drift from actual observed relay performance (based on analysis of Attachments 1 and 3) is:

AD_{Random} = ±0.0552 Vac

AD_{Bias} = 0 Vac

Per Section 6.2.1 of Input 4.13, the Analyzed Drift (AD) characterizes not only the Vendor Drift (VD) but also the Vendor Accuracy (VA) and M&TE (or calibration error). Therefore, it can be conservatively stated that Analyzed Drift (AD) is equal to Vendor Accuracy (VA).

As demonstrated above, Analyzed Drift based on actual observed relay performance is substantially less than the Vendor Accuracy specified by Input 4.8. Since Analyzed Drift is based on actual observed behavior it is considered a more realistic characterization of relay performance. Therefore the Analyzed Drift (AD) will be used for Vendor Accuracy (VA) and also again for Vendor Drift (VD). Refer to Section 2, Methodology, for additional discussion.

A Monticello specific drift analysis of ITE 27N211T4175 relays' voltage function was performed (Attachments 1 and 3) to determine AD.

AD_{Random} = ±0.0552 Vac

AD_{Bias} = 0 Vac

Therefore VA = AD_{Random} + AD_{Bias} = ±0.0552 Vac + 0.0 Vac

Note that the bias term is 0 Vac.

Note 2: Per Input 4.8, the temperature error is characterized as a repeatability error of 0.3 Volts over 72°F range (32°F to 104°F) or 0.00417 Volts/°F. Based on Input 4.14, the Turbine Building Switchgear Rooms have a 44°F temperature range (104°F - 60°F). Therefore, the Accuracy Temperature Effect (ATE) is:

ATE = (0.00417 Vac/°F) x 44°F = 0.1835 Vac

Note 3: Overpressure Effects (OPE) are not applicable to the Degraded Voltage relays.

Note 4: Control power voltage is regulated by the actions of the battery chargers and will not vary significantly from the voltage seen during the quarterly surveillances. Therefore, error effects due to Power Supply Effects (PSE) are included in the VA discussed in Note 1.

Note 5: No RFI/EMI Effects (REE) are identified.

Note 6: Seismic Effects (SE), Radiation Effects (RE), and Humidity Effects (HE) are not specified for these relays. Minor performance variations due to seismic, radiation, or humidity effects would show up in the As Found/As Left data. Therefore, any effects due to these factors are accounted for in the Analyzed Drift, which is being used for the Vendor Accuracy. It should also be noted that the Turbine Building Switchgear Room is not considered to be a harsh environment. Therefore these effects are not considered significant, and Seismic Effects (SE), Radiation Effects (RE), and Humidity Effects (HE) are set to 0.

Note 7: Static Pressure Effects (SPE) are not applicable for electrical devices.

$$A_L = \pm\sqrt{VA^2 + ATE^2 + OPE^2 + SPE^2 + SE^2 + RE^2 + HE^2 + PSE^2 + REE^2}$$

$$A_L = \pm\sqrt{0.0552^2 + 0.1835^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2}$$

$$A_L = \pm 0.1917 \text{ Vac}$$

7.2.1.4 Individual Device Drift:

Term	Value
VD:	Not Specified
DTE:	Not Specified

Vendor Drift (VD) is not specified for the ITE relays. A Monticello specific drift analysis of ITE 27N211T4175 relays' voltage function was performed (Attachments 1 and 3) to determine AD. The AD is used in place of both the VD and the DTE (Drift Temperature Effect).

AD_{Random} = ±0.0552 Vac

AD_{Bias} = 0 Vac

There are no other instruments associated with the Degraded Voltage Relays; therefore, the loop consists of only the relays. Therefore Loop Drift is:

$$D_L = AD_{Random} + AD_{Bias} = \pm 0.0552 \text{ Vac} + 0 \text{ Vac} = \pm 0.0552 \text{ Vac}$$

7.2.1.5 As-Left Tolerance (ALT):

Per the ESM instructions (Section 4.3.3 of Input 4.1), a suggested ALT is determined with the following equation:

$$ALT = \pm\frac{3}{2} \times VA = \pm\frac{3}{2} \times 0.0552 \text{ Vac} = \pm 0.0828 \text{ Vac}$$

Per Input 4.4, the following As Left tolerances are currently being used for these relays:

$$ALT = \pm 0.05 \text{ Vac}$$

The current ALT of ±0.05 is conservative and will be retained.

7.2.1.6 Device Calibration Error:

The calibration procedure, Input 4.4, requires the use of a Datron 1271 digital multimeter (C_1) to read the input at which the trip occurs. Input 4.16 provides the specifications for this multimeter.

Term	Value	Sigma	Reference
C_1:	±0.0353 Vac	3	Note 1
C_{1STD}:	±0.0177 Vac	3	Note 2
ALT:	±0.05 Vac	3	7.2.1.5

Note 1: Per Input 4.16, for the AC Voltage High Accuracy Option 12, the 1000 Volt Range, and the 40 Hz to 10 kHz range, the following accuracy specification is given for a one year calibration. (Note: 115 Volts is conservatively used for reading or setpoint. R is defined as Reading and FS is Full Scale, which is defined as 2 x Range.)

$$Acc = \pm[80 \text{ ppmR} + 10 \text{ ppmFS}]$$

$$Acc = \pm\left[\left(\frac{80}{10^6} \times 115\right) + \left(\frac{10}{10^6} \text{x}2000\right)\right] = \pm 0.0292 \text{ Vac}$$

The Calibration uncertainty is specified as 100 ppm. Therefore,

$$Cal = \pm[100\ ppmR]$$
$$Cal = \pm\left[\left(\frac{100}{10^6}\times 115\right)\right] = \pm 0.0115\ Vac$$

The Temperature Effect is specified as 10 ppm/°C. The calibration temperature range is 65°F to 90°F.

$$TE = \pm\left[\frac{10\ ppmR}{°C}\times\frac{1°C}{1.8°F}\times(90°F - 65°F)\times 115\right]$$
$$TE = \pm\left[\left(\frac{10}{10^6}\right)\times\frac{1°C}{1.8°F}\times 25°F\times 115\right] = \pm 0.0160\ Vac$$

Combination of the above terms yields C_1.

$$C_1 = \pm\sqrt{Acc^2 + Cal^2 + TE^2}$$
$$C_1 = \pm\sqrt{0.0292^2 + 0.0115^2 + 0.0160^2}$$
$$C_1 = \pm 0.0353\ Vac$$

Note 2: The subject meter is not calibrated in-house but is shipped offsite for calibration. Reference 5.6 is an example calibration certificate. The certificate makes the following statement, "All services provided a 4:1 uncertainty ratio (or greater) unless otherwise stated." Due to the varying range of uncertainties of the instrument, based on range, it is difficult to ensure with 100% confidence that a full 4:1 uncertainty ratio is maintained. Therefore, a conservative ratio of 2:1 will be considered to determine the calibration standard error (Assumption 6.3). Therefore,

$$C_{1STD} = \frac{C_1}{2} = \pm\frac{0.0353}{2} = \pm 0.0177\ Vac$$

Since calibration term values are controlled by 100% testing, they represent 3-sigma values. Individual calibration error terms are combined using the SRSS method and normalized to a 2-sigma confidence level.

$$C_L = \pm\frac{2}{3}\sqrt{C_1^2 + C_{1STD}^2 + ALT^2}$$

$$C_L = \pm\frac{2}{3}\sqrt{0.0353^2 + 0.0177^2 + 0.05^2}$$

$$C_L = \pm 0.0425 \text{ Vac}$$

7.3 Determination of Primary Element Accuracy (PEA) and Process Measurement Accuracy (PMA) – Voltage Function

There are no PMA inaccuracies associated with the Degraded Voltage function.

$$PMA = 0$$

Per Input 4.5, the technical manual, the Potential Transformers for these circuits are produced in accordance with American Standards for Instrument Transformers, ASA C57.13. Input 4.6 is the current version of that standard. Per Section 5 of the standard, "Accuracy Classes for Metering Service," Table 6 lists the metering accuracy classes as 0.3%, 0.6%, and 1.2%. Per Work Order 0401473 (Input 4.24), the Bus 15 transformer is a Type JVM-3 Potential Transformer. At the time of the preparation of this calculation, the Bus 16 transformer is not available for inspection. Per Assumption 6.1, the Bus 16 transformer is assumed to be the same as Bus 15. Per Attachment 8 (from Input 4.22), the JVM-3 designation confirms that the transformer has an accuracy class of 0.3%, with a variable standard burden designation up to Y. Attachment 8 contains a characteristic curve of General Electric Type JVM-3 and JVW-3 potential transformers (from Input 4.23). From Attachment 8, the apex of the fan curve (which corresponds to zero burden) for this type of transformer originates at a ratio correction factor of approximately 0.9975. The standard burden curve for designation M is considered conservative for this application because the listing of loads for this transformer shows that the normal loading is approximately 25.5 VA, per Attachment 9. The M, which indicates a burden of 35 VA per Attachment 8, is considered conservatively high on the projected burden, and therefore provides an upper limit for the ratio correction factor. The burden designation of M approaches a correction factor of 1.000, but never exceeds it. Therefore, the PEA error is considered to be a negative bias for this application, which is maximized at a burden of zero for the transformer. The maximum bias value is computed conservatively, based on the upper Analytical Limit voltage.

$$PEA_{Bias} = (0.9975 - 1) x (112.3714 \text{ Vac}) = -0.2810 \text{ Vac}$$

The error is defined at a burden of zero, which means that the output voltage from the transformer is higher than ideal for a given input voltage. This means that on a decreasing voltage trend, the trip will take longer than it would have without the error. Therefore, this is equivalent to a negative shift in the setpoint.

7.4 Determination of Other Error Terms – Voltage Function

No other errors are applicable to the Degraded Voltage function.

7.5 Calculation of Allowable Value and Operating Setpoint – Voltage Function

7.5.1 Allowable Value (AV):

Per Input 4.2, the Current Technical Specifications provide ± limits on the Degraded Voltage setting, thus establishing two Allowable Values. Per Section 7.1, the function of the Degraded Voltage relay is to provide a transfer to onsite power sources in the event offsite grid voltage declines to a sustained level such that, under maximum load conditions, the offsite grid voltage does not provide the capability to start and run all Class 1E equipment within the equipment voltage ratings.

Per the AC Electrical Load Study (Input 4.11 – Appendix II Note 2), the Degraded Voltage limits are established as follows:

The upper Degraded Voltage limit for motor starting studies is the Technical Specification Setpoint plus the high side setting tolerance:

3915 + 18 = 3933 Vac (Bus Voltage);
Converting to relay voltage yields:
3933/35 = 112.3714 Vac

The lower Degraded Voltage limit for motor starting studies is the Technical Specification Setpoint minus the low side setting tolerance:

3915 - 18 = 3897 Vac (Bus Voltage);
Converting to relay voltage yields:
3897/35 = 111.3429 Vac

Using these limits as the Lower and Upper Analytical Limits yields:

	References
Upper Analytical Limit (AL_U): ≤ 112.3714 Vac	Input 4.11
Lower Analytical Limit (AL_L): ≥ 111.3429 Vac	Input 4.11

The Allowable Values can now be computed:

Term	Value (Vac) (relay voltage)	Sigma	Reference
A_L	±0.1917	2	Section 7.2.1.3
C_L	±0.0425	2	Section 7.2.1.6
PMA	0.0000	2	Section 7.3
PEA_{Bias}	-0.2810	NA	Section 7.3

The negative PEA_{Bias} term is applied to the lower value, since it acts to lower the setpoint. It is in the conservative direction for the upper setpoint, and is therefore not considered.

$$AV_U \le AL_U - \frac{1.645}{2}\left(\sqrt{A_L^2 + C_L^2 + PMA^2}\right) - \text{bias terms}$$

$$AV_U \le 112.3714 - \frac{1.645}{2}\left(\sqrt{0.1917^2 + 0.0425^2 + 0.0000^2}\right) - 0$$

$$AV_U \le 112.3714 - 0.1616 - 0$$

$$AV_U \le 112.2098 \text{ Vac}$$

$$AV_U \text{ (Bus Voltage)} \le 112.2098 \times 35 \le 3927.343 \text{ Vac Bus Voltage}$$

$$AV_U \text{ (Bus Voltage)} \le 3927 \text{ Vac Bus Voltage (after rounding)}$$

$$AV_L \ge AL_L + \frac{1.645}{2}\left(\sqrt{A_L^2 + C_L^2 + PMA^2}\right) - PEA_{Bias}$$

$$AV_L \ge 111.3429 + \frac{1.645}{2}\left(\sqrt{0.1917^2 + 0.0425^2 + 0.000^2}\right) - (-0.2810)$$

$$AV_L \ge 111.3429 + 0.1616 + 0.2810$$

$$AV_L \ge 111.7855 \text{ Vac}$$

$$AV_L \text{ (Bus Voltage)} \ge 111.7855 \times 35 \ge 3912.493 \text{ Vac Bus Voltage}$$

$$AV_L \text{ (Bus Voltage)} \ge 3913 \text{ Vac Bus Voltage (after rounding)}$$

Currently, per Input 4.2, the allowed limits from the Technical Specification Deviation Table are ≥ 3897 Vac Bus Voltage (trip) and ≤3975 Vac Bus Voltage (reset). The calculated lower limit Allowable Value is more conservative than the ≥ 3897 Vac Bus Voltage (trip) value.

AV_U ≤ 3927 Vac Bus Voltage;

AV_L ≥ 3913 Vac Bus Voltage

These values should be used as the Improved Technical Specifications Allowable Values.

7.5.2 Nominal Trip Setpoint ($NTSP_1$):

The Nominal Trip Setpoint ($NTSP_1$) can now be computed:

Term	Value (Vac) (relay voltage)	Sigma	Reference
A_L	±0.1917	2	Section 7.2.1.3
AD_{Random}	±0.0552	2	Section 7.2.1.4
C_L	±0.0425	2	Section 7.2.1.6
PMA	0.0000	2	Section 7.3
PEA_{Bias}	-0.2810	NA	Section 7.3

The negative PEA_{Bias} term is applied to the lower value, since it acts to lower the setpoint. It is in the conservative direction for the upper setpoint, and is therefore not considered.

$$NTSP_{1U} \le AL_U - \frac{1.645}{2}\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random} + PMA^2}\right) - bias$$

$$NTSP_{1U} \le 112.3714 - \frac{1.645}{2}\left(\sqrt{0.1917^2 + 0.0425^2 + 0.0552^2 + 0.0000^2}\right) - 0$$

$$NTSP_{1U} \le 112.3714 - 0.1678 - 0$$

$$NTSP_{1U} \le 112.2036 \text{ Vac}$$

$$NTSP_{1L} \ge AL_L + \frac{1.645}{2}\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random} + PMA^2}\right) - PEA_{Bias}$$

$$NTSP_{1L} \ge 111.3429 + \frac{1.645}{2}\left(\sqrt{0.1917^2 + 0.0425^2 + 0.0552^2 + 0.0000^2}\right) - (-0.2810)$$

$$NTSP_{1L} \ge 111.3429 + 0.1678 + 0.2810$$

$$NTSP_{1L} \ge 111.7917 \text{ Vac}$$

Therefore, the $NTSP_1$ are:

$NTSP_{1U} \le 112.2036$ Vac
$NTSP_{1L} \ge 111.7917$ Vac

7.5.3 Licensee Event Report (LER) Avoidance Evaluation:

The purpose of the LER Avoidance Evaluation is to assure that there is sufficient margin provided between the AV and the NTSP to reasonably avoid violations of the AV. Any Z value greater than 1.29 provides sufficient margin between the NTSP and the AV. Therefore, $NTSP_2$ is calculated to provide bounds for the NTSP based on LER avoidance criteria.

$$\text{Sigma(LER)} = \left(\frac{1}{2}\right)\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random}{}^2}\right)$$

$$\text{Sigma(LER)} = \left(\frac{1}{2}\right)\left(\sqrt{0.1917^2 + 0.0425^2 + 0.0552^2}\right)$$

$$\text{Sigma(LER)} = 0.1020 \text{ Vac}$$

$$NTSP_{2U} \leq AV_U - (Z \times \text{Sigma(LER)}) + D_{U.Bias}$$

$$NTSP_{2U} \leq 112.2098 - (1.29 \times 0.1020) + 0$$

$$NTSP_{2U} \leq 112.0782 \text{ Vac}$$

$$NTSP_{2L} \geq AV_L + (Z \times \text{Sigma(LER)}) + D_{L.Bias}$$

$$NTSP_{2L} \geq 111.7855 + (1.29 \times 0.1020) + 0$$

$$NTSP_{2L} \geq 111.9171 \text{ Vac}$$

Therefore, an $NTSP_2 \geq 111.9171$ Vac and ≤ 112.0782 Vac results in a Z greater than 1.29 and provides sufficient margin between the NTSP and the Allowable Values.

7.5.4 Selection of Operating Setpoint:

Per Section 5.6.4 of Input 4.1, the operating setpoint of 111.9977 Vac, rounded to 112.00 Vac, is chosen between the $NTSP_2$ limits.

NTSP = 112.00 Vac; or 35 x 112.00 Vac = 3920 Vac Bus Voltage;

Note that the NTSP of 112.00 combined with the ALT of ±0.05 Vac is within the limits of $NTSP_{2U}$ and $NTSP_{2L}$. Therefore, the Allowable Values are protected so the NTSP is acceptable.

7.5.5 Leave Alone Zone:

Leave Alone Zones/Tolerances as described in the GE documents are not used at Monticello Plant.

7.5.6 Establishing As-Found Tolerance (AFT):

The AFT is calculated to provide limits for use during surveillance testing. The Analyzed Drift term is used in place of the VD and DTE terms:

$$AFT = \pm\sqrt{VA^2 + VD^2 + DTE^2 + C_L{}^2}$$

$$AFT = \pm\sqrt{0.0552^2 + 0.0552^2 + 0.0177^2} = \pm 0.08 \text{ Vac}$$

The AFT value is greater than the ALT of ±0.05 Vac, and when applied to the setpoint of 112.00 Vac, provides a band of 111.92 to 112.08 Vac. This band is well within the Allowable Value band of 111.7855 to 112.2098 Vac, and is therefore acceptable.

7.5.7 Required Limits Evaluation:

The purpose of a Required Limits Evaluation is to assure that the combination of errors present during calibration of each device in the channel is accounted for while allowing for the possibility that the devices may not be recalibrated. Since Leave Alone Zones are not used at MNGP, the devices are always verified or recalibrated to be within the As Left Zone. Therefore, a Required Limits Evaluation as discussed in the GE methodology is not applicable.

7.5.8 Spurious Trip Avoidance Evaluation:

The purpose of a spurious trip avoidance evaluation is to assure that there is a reasonable probability that spurious trips will not occur using the selected setpoint. The Upper Allowable Value and Setpoint evaluations in this calculation are performed to ensure that potential transients on the grid system and voltage drop due to starting of large motors would not cause spurious trips. Therefore, no separate evaluation is necessary.

7.5.9 Elevation Correction:

Not applicable.

7.5.10 Determination of Actual Setpoint / Instrument Scaling:

The setpoint of 112.00 Vac, or 3920 Vac Bus Voltage is used. Note that a loop scaling factor of 35/1 is applicable between the relay setting and the bus voltage. Attachment 5 is a Setpoint Relationship Diagram for the Degraded Voltage Relay Voltage Function.

7.6 Instrument Definition and Determination of Device Error Terms – Time Delay Function

7.6.2 DEVICE 2

7.6.2.1 Instrument Definition:

		Reference
Component ID:	**127-5A, -5B, -5C**	
Location:	Turbine Building, Elevation 911', Lower Level, Lower 4KV Room, Cubicle 152-510	4.3
Manufacturer:	ITE	4.12
Model Number:	27N211T4175	4.12
Setpoint:	9.0 seconds	4.4
Output Signal:	Contact Output	4.4

		Reference
Component ID:	**127-6A, -6B, -6C**	
Location:	Turbine Building, Elevation 931', Lower Level, Ground Floor 4KV Room, Cubicle 152-601	4.3
Manufacturer:	ITE	4.12
Model Number:	27N211T4175	4.12
Setpoint:	9.0 seconds	4.4
Output Signal:	Contact Output	4.4

7.6.2.2 Process and Physical Interfaces:

Calibration Conditions:		Reference
Temperature:	65 to 90°F	5.1
Current Surveillance Interval for Loss of Voltage Relays:	Quarterly	4.2
Proposed Surveillance Interval for Degraded Voltage Relays:	Quarterly	Note: The Degraded Voltage Relay

		calibration interval is not being extended based on this calculation.

Normal / Trip Plant Environmental Conditions:		Reference
Average Temperature:	60°F to 104°F	4.14

Seismic Conditions:		Reference
OBE Prior to Function:	N/A	N/A
OBE During Function:	N/A	N/A

These relays respond to a degraded voltage condition that is not related to any DBA or seismic event. Therefore, seismic conditions are not required to be determined for the Degraded Voltage relays.

Process Conditions:		Reference
During Calibration:	N/A	N/A
Worst Case:	N/A	N/A
During Function:	N/A	N/A

During the event when these devices are required, the Degraded Voltage relays are not subjected to process conditions (static pressure, overpressure, elevated temperatures, etc.) that would affect the accuracy of the instrument.

7.6.2.3 Individual Device Accuracy:

Term	Value	Sigma	Reference
VA:	±0.020 Seconds	2	Input 4.8; Note 1
ATE:	0	N/A	Note 2
OPE:	NA	N/A	Note 3
SPE:	NA	N/A	Note 6
SE:	0	N/A	Note 5
RE:	0	N/A	Note 5
HE:	0	N/A	Note 5
PSE:	NA	N/A	Note 4
REE:	NA	N/A	Note 4

Note 1: Per Input 4.8, the Vendor Accuracy is:
VA = ± 0.020 Seconds.

Note 2: Per Input 4.8, there is not an Accuracy Temperature Effect (ATE) specified. Therefore, any temperature effect is encompassed in the Vendor Accuracy, which is included in the Analyzed Drift data. These relays are not located in an area with extreme temperature variations – therefore any temperature effect would be reflected in the Analyzed Drift term. Therefore no additional uncertainty is applied due to ATE.

Note 3: Overpressure Effects (OPE) are not applicable to the Degraded Voltage relays.

Note 4: No error effects due to Power Supply Effects (PSE) and RFI/EMI Effects (REE) are identified.

Note 5: Seismic Effects (SE), Radiation Effects (RE), and Humidity Effects (HE) are not specified for these relays. Minor performance variations due to seismic, radiation, or humidity effects would show up in the As Found/As Left data. Therefore, any effects due to these factors are accounted for in the Analyzed Drift, which is being used for the Vendor Accuracy. It should also be noted that the Turbine Building Switchgear Room is not considered to be a harsh environment. Therefore these effects are not considered significant, and Seismic Effects (SE), Radiation Effects (RE), and Humidity Effects (HE) are set to 0.

Note 6: Static Pressure Effects (SPE) do not apply to time delay devices (Reference 5.1).

$$A_L = \pm\sqrt{VA^2 + ATE^2 + OPE^2 + SPE^2 + SE^2 + RE^2 + HE^2 + PSE^2 + REE^2}$$
$$A_L = \pm\sqrt{0.020^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2}$$
$$A_L = \pm 0.020 \text{ Seconds}$$

7.6.2.4 Individual Device Drift:

Term	Value
VD:	Not Specified
DTE:	Not Specified

Vendor Drift (VD) is not specified for the relays. A Monticello specific drift analysis of ITE 27N211T4175 relays' time delay function was performed (Attachments 2 and 4) to determine AD. The AD is used in place of both the VD and the DTE (Drift Temperature Effect).

AD_{Random} = ±0.0801 Seconds

AD_{Bias} = 0 Seconds

There are no other instruments associated with the Degraded Voltage Relays, therefore, the loop consists of only the relays. Therefore Loop Drift is:

$D_L = AD_{Random} + AD_{Bias}$ = ±0.0801 Seconds + 0 Seconds = ±0.0801 Seconds

7.6.2.5 As-Left Tolerance (ALT):

Per the ESM instructions (Section 4.3.3 of Input 4.1), a suggested ALT is determined with the following equation:

$$ALT = \pm\frac{3}{2} \times VA = \pm\frac{3}{2} \times 0.0801 \text{ Seconds} = \pm 0.1202 \text{ Seconds}$$

Per Input 4.4, the following As Left tolerances are currently being used for these relays, and will be retained.

ALT = ±0.10 Seconds

7.6.2.6 Device Calibration Error:

Term	Value	Sigma	Reference
C_1:	±0.0200 Seconds	3	Note 1
C_{1STD}:	±0.0200 Seconds	3	Note 2
ALT:	±0.1000 Seconds	3	7.6.2.5

Note 1: The Calibration Tool Error (C_1) is considered equal to the accuracy of the timer used to calibrate the EPA timer. Per Input 4.4, this is a Doble 2000 Series variable voltage / frequency test equipment (or equivalent) with Controller. Per Input 4.21, the accuracy of this device is ±0.01% of reading and ± 1 L.S.D. For a nominal setpoint of 9 seconds, this produces the following accuracy value.

$$C_1 = \pm(0.01\% \times \text{Reading} + 1\,\text{LSD})$$
$$C_1 = \pm(0.0001 \times 9\ \text{seconds} + 0.01\ \text{seconds})$$
$$C_1 = \pm 0.0109\ \text{seconds}$$
$$C_1 = \pm 0.02\ \text{seconds (conservatively rounded)}$$

Note 2: In accordance with Input 4.1, the calibration standard error (C_{1STD}) is considered to be equal to C_1.

Since calibration term values are controlled by 100% testing, they represent 3-sigma values. Individual calibration error terms are combined using the SRSS method and normalized to a 2-sigma confidence level.

$$C_L = \pm\frac{2}{3}\sqrt{C_1^2 + C_{1STD}^2 + ALT^2}$$
$$C_L = \pm\frac{2}{3}\sqrt{0.0200^2 + 0.0200^2 + 0.1000^2}$$
$$C_L = \pm 0.0693\ \text{Seconds}$$

7.7 Determination of Primary Element Accuracy (PEA) and Process Measurement Accuracy (PMA) – Time Delay Function

There are no PMA inaccuracies associated with the time delay function.

PMA = 0

There are no PEA inaccuracies associated with the time delay function.

PEA = 0

7.8 Determination of Other Error Terms – Time Delay Function

No other errors are applicable to the Degraded Voltage time delay function.

7.9 Calculation of Allowable Value and Operating Setpoint – Time Delay Function

7.9.1 Allowable Value (AV):

Per Input 4.2, the Current Technical Specifications provide ± limits on the time delay setting, thus establishing two Allowable Values. Per Section 7.1, the function of the Degraded Voltage relay time delay function is to provide a transfer to onsite power sources in the event offsite grid voltage declines to a sustained level such that, under maximum load conditions, the offsite grid voltage does not provide the capability to start and run all Class 1E equipment within the equipment voltage ratings.

The Degraded Voltage time delay setpoint and setting tolerance established in the plant Technical Specifications is:

9 ± 1 Seconds

Input 4.15 provides the basis for the 10 second upper limit. Section 2.4.2 of Input 4.15 requires a bus transfer on a degraded voltage condition in less than or equal to 10 seconds. The 8 second lower limit is designed to minimize or prevent the transfer during short voltage transients.

Using these limits as the Lower and Upper Analytical Limits yields:

Lower Analytical Limit (AL_L): ≥ 8 Sec
Upper Analytical Limit (AL_U): ≤ 10 Sec

In order to maintain the current field setting of 9 Seconds a margin of ±0.7407 Seconds is included (although not required) in the computation of the Allowable Value and the Nominal Trip Setpoint ($NTSP_1$).

The Allowable Values can now be computed:

Term	Value (Seconds)	Sigma	Reference
A_L	±0.0200	2	Section 7.6.2.3
C_L	±0.0693	2	Section 7.6.2.6
PMA	0	2	Section 7.7
PEA	0	2	Section 7.7
Margin	±0.7407	2	Section 7.9.1

$$AV_U \leq AL_L - \frac{1.645}{2}\left(\sqrt{A_L^2 + C_L^2 + PMA^2 + PEA^2}\right) - \text{margin} - \text{bias terms}$$

$$AV_U \leq 10.0000 - \frac{1.645}{2}\left(\sqrt{0.0200^2 + 0.0693^2 + 0.0000^2 + 0.0000^2}\right) - 0.7407 - 0$$

$$AV_U \leq 10.0000 - 0.0593 - 0.7407 + 0$$

$$AV_U \leq 9.20 \text{ Seconds}$$

$$AV_L \geq AL_L + \frac{1.645}{2}\left(\sqrt{A_L{}^2 + C_L{}^2 + PMA^2 + PEA^2}\right) + \text{margin} + \text{bias terms}$$

$$AV_L \geq 8.0000 + \frac{1.645}{2}\left(\sqrt{0.0200^2 + 0.0693^2 + 0.0000^2 + 0.0000^2}\right) + 0.7407 - 0$$

$$AV_L \geq 8.0000 + 0.0593 + 0.7407 + 0$$

$$AV_L \geq 8.80 \text{ Seconds}$$

Currently, per Input 4.2, the allowed limits from the Technical Specification Deviation Table are ≥ 5 Seconds and ≤ 10 Seconds. The calculated Allowable Values are more conservative than the ≥ 5 Seconds and ≤ 10 Seconds.

AV_U = 9.20 Seconds;

AV_L = 8.80 Seconds

These values should be used as the Improved Technical Specifications Allowable Value.

7.9.2 <u>Nominal Trip Setpoint ($NTSP_1$):</u>

The Nominal Trip Setpoint ($NTSP_1$) can now be computed:

Term	Value (Seconds)	Sigma	Reference
A_L	±0.0200	2	Section 7.6.2.3
AD_{Random}	±0.0801	2	Section 7.6.2.4
C_L	±0.0693	2	Section 7.6.2.6
PMA	0.0000	2	Section 7.7
PEA	0.0000	2	Section 7.7
Margin	±0.7407	2	Section 7.9.1

$$NTSP_{1U} \leq AL_U - \frac{1.645}{2}\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random} + PMA^2 + PEA^2}\right) - \text{margin} - \text{bias}$$

$$NTSP_{1U} \leq 10.0000 - \frac{1.645}{2}\left(\sqrt{0.0200^2 + 0.0693^2 + 0.0801^2 + 0.0000^2 + 0.0000^2}\right) - 0.7407 - 0$$

$$NTSP_{1U} \leq 10.0000 - 0.0886 - 0.7407 - 0$$

$$NTSP_{1U} \leq 9.1707 \text{ Seconds}$$

$$NTSP_{1L} \geq AL_L + \frac{1.645}{2}\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random} + PMA^2 + PEA^2}\right) + \text{margin} + \text{bias}$$

$$NTSP_{1L} \geq 8.0000 + \frac{1.645}{2}\left(\sqrt{0.0200^2 + 0.0693^2 + 0.0801^2 + 0.0000^2 + 0.0000^2}\right) + 0.7407 + 0$$

$$NTSP_{1L} \geq 8.0000 + 0.0886 + 0.7407 + 0$$

$$NTSP_{1L} \geq 8.8293 \text{ Seconds}$$

Therefore, the Nominal Trip Setpoints are:

$NTSP_{1U}$ = 9.1707 Seconds
$NTSP_{1L}$ = 8.8293 Seconds

7.9.3 Licensee Event Report (LER) Avoidance Evaluation:

The purpose of the LER Avoidance Evaluation is to assure that there is sufficient margin provided between the AV and the NTSP to reasonably avoid violations of the AV. Any Z value greater than 1.29 provides sufficient margin between the NTSP and the AV. Therefore, $NTSP_2$ is calculated to provide bounds for the NTSP based on LER avoidance criteria.

$$\text{Sigma(LER)} = \left(\frac{1}{2}\right)\left(\sqrt{A_L{}^2 + C_L{}^2 + AD_{Random}{}^2}\right)$$

$$\text{Sigma(LER)} = \left(\frac{1}{2}\right)\left(\sqrt{0.0200^2 + 0.0693^2 + 0.0801^2}\right)$$

$$\text{Sigma(LER)} = 0.0539 \text{ Seconds}$$

$$NTSP_{2U} = AV_U - (Z \times \text{Sigma(LER)}) - D_{U.Bias}$$

$$NTSP_{2U} = 9.20 - (1.29 \times 0.0539) - 0$$

$$NTSP_{2U} = 9.1305 \text{ Seconds}$$

$$NTSP_{2L} = AV_L + (Z \times \text{Sigma(LER)}) + D_{L.Bias}$$

$$NTSP_{2L} = 8.80 + (1.29 \times 0.0539) + 0$$

$$NTSP_{2L} = 8.8695 \text{ Seconds}$$

Therefore, an $NTSP_2 \geq 8.8695$ Seconds and ≤ 9.1305 Seconds results in a Z greater than 1.29 and provides sufficient margin between the NTSP and the Allowable Values.

7.9.4 Selection of Operating Setpoint:

Per Section 5.6.4 of Input 4.1, the operating setpoint of 9.0 Seconds is chosen between the $NTSP_2$ limits. This value will be the NTSP:

NTSP = 9.0 Seconds; Therefore the current time delay setting is retained.

7.9.5 Leave Alone Zone:

Leave Alone Zones/Tolerances as described in the GE documents are not used at Monticello Plant.

7.9.6 Establishing As-Found Tolerance (AFT):

The AFT is calculated to provide limits for use during surveillance testing. The Analyzed Drift term is used in place of the VD and DTE terms:

$$AFT = \pm\sqrt{VA^2 + VD^2 + DTE^2 + C_L{}^2}$$

$$AFT = \pm\sqrt{0.02^2 + 0.08^2 + 0.0693^2} = \pm 0.11 \cong \pm 0.20 \text{ seconds}$$

The AFT value is greater than the ALT of ±0.10 Seconds, and when applied to the setpoint of 9 Seconds, provides a band of 8.8 to 9.2 Seconds. This band is equal to the Allowable Value and is therefore acceptable.

7.9.7 Required Limits Evaluation:

The purpose of a Required Limits Evaluation is to assure that the combination of errors present during calibration of each device in the channel is accounted for while allowing for the possibility that the devices may not be recalibrated. Since Leave Alone Zones are not used at MNGP, the devices are always verified or recalibrated to be within the As Left Zone. Therefore, a Required Limits Evaluation as discussed in the GE methodology is not applicable.

7.9.8 Spurious Trip Avoidance Evaluation:

The purpose of a spurious trip avoidance evaluation is to assure that there is a reasonable probability that spurious trips will not occur using the selected setpoint. The Upper Allowable Value and Setpoint evaluations in this calculation are performed to ensure that potential transients on the grid system and voltage drop due to starting of large motors would not cause spurious trips. Therefore, no separate evaluation is necessary.

7.9.9 Elevation Correction:

Not applicable.

7.9.10 Determination of Actual Setpoint / Instrument Scaling:

The setpoint of 9.0 Seconds is used. No conversions of units are required for loop scaling purposes. Attachment 6 is a Setpoint Relationship Diagram for the Degraded Voltage Relay Time Delay Function.

8. CONCLUSIONS

Voltage Function calculation results:

Terms	Relay Voltage (Vac)	Bus Voltage (Vac)	Section
A_L:	±0.1917	±6.710	7.2.1.3
AD_{Random}:	±0.0552	±1.932	7.2.1.4
D_{Bias}:	0	0	7.2.1.4
ALT:	±0.050	±1.75	7.2.1.5
C_L:	±0.0425	±1.4875	7.2.1.6
PEA_{Bias}:	-0.2810	-9.835	7.3
PMA:	0	0	7.3
AL_L:	≥ 111.3429	≥ 3897	7.5.1
AL_U:	≤ 112.3714	≤ 3933	7.5.1
AV_L:	≥ 111.7855	≥ 3913	7.5.1
AV_U:	≤ 112.2098	≤ 3927	7.5.1
Current Technical Specification Trip Setting:	111.86 ±0.51	3915 ±18	4.2
Current Trip Setpoint:	111.96	3918.6	4.4
Proposed Trip Setpoint:	112.00	3920	7.5.4
$NTSP_{1L}$:	≥ 111.7917	≥ 3912.71	7.5.2
$NTSP_{1U}$:	≤ 112.2036	≤ 3927.12	7.5.2
$NTSP_{2L}$:	≥ 111.9171	≥ 3917.10	7.5.3
$NTSP_{2U}$:	≤ 112.0782	≤ 3922.73	7.5.3
AFT:	±0.08	±2.8	7.5.6
AF Upper Limit:	≤ 112.08	≤ 3922.8	7.5.6

AF Lower Limit:	≥ 111.92	≥ 3917.2	7.5.6
Elevation Correction:	NA	NA	7.5.9

The negative PEA_{Bias} term is due to the Potential Transformer, and acts in such a way to decrease the setpoint for the degraded voltage function.

For the Degraded Voltage Relays Voltage Function this calculation determined the following Allowable Values for use in the MNGP Improved Technical Specifications:

AV_L: ≥ 3913 Vac (Bus Voltage)
AV_U: ≤ 3927 Vac (Bus Voltage)

This calculation determines a new setpoint of 112.00 Vac (Relay Voltage)/3920 (Bus Voltage). The As Left Tolerance remains unchanged at ± 0.05 Vac (Relay Voltage). Following approval of the ITS Amendment request, the AFT will be changed to ±0.08 Vac (Relay Voltage).

Time Delay calculation results:

Terms	Time Delay (Seconds)	Section
A_L:	±0.0200	7.6.2.3
AD_{Random}:	±0.0801	7.6.2.4
D_{Bias}:	0	7.6.2.4
ALT:	±0.1000	7.6.2.5
C_L:	±0.0693	7.6.2.6
PEA:	±0.0000	7.7
PMA:	±0.0000	7.7
AL_L:	≥ 8.0	7.9.1
AL_U:	≤ 10.0	7.9.1
AV_L:	≥ 8.80	7.9.1
AV_U:	≤ 9.20	7.9.1
Current Technical Specification Trip Setting	9.0	7.9.4
Proposed Trip Setting	9.0	7.9.4
$NTSP_{1L}$:	≥ 8.8293	7.9.2
$NTSP_{1U}$:	≤ 9.1707	7.9.2
$NTSP_{2L}$:	≥ 8.8695	7.9.3

$NTSP_{2U}$:	≤ 9.1305	7.9.3
AFT:	±0.20	7.9.6
AF Lower Limit:	≥ 8.80	7.9.6
AF Upper Limit:	≤ 9.20	7.9.6
Elevation Correction:	NA	7.9.9

For the Degraded Voltage Relays Time Delay function this calculation determined the following Allowable Values for use in the MNGP Improved Technical Specifications:

AV_L: ≥ 8.80 Seconds
AV_U: ≤ 9.20 Seconds

The current setpoint of 9.0 Seconds and As Left Tolerance ± 0.10 Seconds do not change. Following approval of the ITS Amendment request, the AFT will be changed to ±0.20 Seconds.

9. FUTURE NEEDS

9.1 Process Setpoint Change Request to implement the following changes/additions for the Degraded Voltage Relay Voltage and Time Delay Functions following approval of the ITS license amendment (AR 00824816-01):

Voltage Function:

1. As Found Tolerance of ±0.08 Vac.
2. As Left Tolerance to ±0.05 Vac.
3. Allowable Value (Lower/Upper) of ≥3913 Vac /≤3927 Vac (Bus Voltage)

Time Delay Function:

1. As Found Tolerance of ±0.20 Seconds.
2. As Left Tolerance to ±0.10 Seconds.
3. Allowable Value (Lower/Upper) of ≥8.80 Seconds/≤9.20 Seconds.

9.2 For the Degraded Voltage Relay Voltage and Time Delay Functions include the following in the Improved Technical Specifications License Amendment Request (AR 00628275-01).

Voltage Function:
Allowable Value (Lower/Upper) of ≥3913 Vac/≤3927 Vac (Bus Voltage)

Time Delay Function:
Allowable Value (Lower/Upper) of ≥8.80 Seconds/≤9.20 Seconds

9.3 Perform a walkdown of the instrument Potential Transformers BUS-16/POT to confirm that the Bus-16 transformer is a Type JVM-3; see Assumption 6.1 (AR 00707981, Work Order 00135892).

9.4 Revise MWI-3-M-2.01 (Input 4.11) to include that Calculation CA-92-220 uses the load study degraded voltage limits of ≥ 3897 volts and ≤ 3933 volts as the Analytical Limit in determining the actual plant setpoints (AR 01018333).

9.5 Revise USAR Sections 8.4.1.3 and 8.10 to reflect change in nominal trip setting (AR 01019846).

10. ATTACHMENTS

1. Instrument Drift Analysis ITE-27N Undervoltage Relays – 4.16 KV Degraded Voltage (Voltage Function).
2. Instrument Drift Analysis ITE-27N Undervoltage Relays – 4.16 KV Degraded Voltage (Time Delay Function).
3. Instrument Drift Analysis Computation Spreadsheet ITE-27N Undervoltage Relay – 4.16 KV Degraded Voltage (Voltage Function).
4. Instrument Drift Analysis Computation Spreadsheet ITE-27N Undervoltage Relay – 4.16 KV Degraded Voltage (Time Delay Function).
5. Setpoint Relationship Diagram (Voltage Function).
6. Setpoint Relationship Diagram (Time Delay Function).
7. Excerpt from Doble F2000 Operating Manual
8. Technical Information for GE Type JVM-3 Potential Transformers.
9. Loading Determination for Potential Transformer
10. Yokogawa Switchboard Instruments Specifications

Close

NRC ITS TRACKING

NRC Reviewer

ID	200510281243 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3; ITS Number: BSI 1a; DOC Number: L.12; JFD Number: None; Bases JFD Number: None; Page Number(s): 7
Comment	This is a Beyond Scope Issue (TAC No. MC7597) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 12/15/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
Licensee Response by Jerry Jones on 03/12/2006	CTS Table 3.1.1 provides trip settings for Function 3.a (Neutron Flux Intermediate Range Monitor (IRM) - High High), Function 4.a (Flow Referenced Neutron Flux Average Power Range Monitor (APRM) High High), and Function 4.c (Flow Referenced Neutron Flux APRM - High Flow Clamp). These trip settings are being replaced with Allowable Values to be consistent with the format of the Reactor Protection System instrumentation table within NUREG-1433. These Allowable Values are therefore consistent with the current licensing basis and the setpoint methodology used to establish the trip settings.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   a. CTS Table 3.1.1 Function 3.a (Improved Technical Specification (ITS) Table 3.3.1.1-1, Function 1.a), Neutron Flux Intermediate Range Monitor (IRM) - High-High, was previously considered an LSSS.

      The IRMs monitor neutron flux levels from the upper range of the source range monitor (SRM) to the lower range of the average power range monitors (APRMs). The IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power range. In this power range, the most significant source of reactivity change is due to control rod withdrawal. The IRM provides diverse protection for the rod worth minimizer (RWM), which monitors and controls the movement of control rods at low power. The RWM prevents the withdrawal of an out of sequence control rod during startup that could result in an unacceptable neutron flux excursion and the IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. This analysis, which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel energy depositions below the 170 cal/gm fuel failure threshold criterion. This was the analysis that supported the IRMs being LSSS, as was in the CTS prior to Amendment 128. However, the current plant-specific control rod withdrawal error analysis does not credit the IRM - High High Function. The plant-specific analysis assumes an out of sequence rod is withdrawn (i.e., RWM does not preclude the event). Results from the analysis show that the core power peak is limited to less than 25% Rated Thermal Power (RTP) and no power distribution limit is exceeded. The analysis does not assume the IRMs mitigate the control rod withdrawal event since the Doppler feedback effect turns the reactor power excursion before the IRM Allowable Value (20% RTP) is reached. In addition, while the IRMs are capable of limiting other reactivity excursions during startup, such as cold water injection events, no credit is specifically assumed. Therefore, the IRM - High High Function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for being an LSSS.

b. CTS Table 3.1.1 Functions 4.a and 4.c (ITS Table 3.3.1.1-1, Function 2.a), Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High and Flow Referenced Neutron Flux APRM - High Flow Clamp, are an LSSS at Monticello.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200510281245** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.3** DOC Number: **L.4** JFD Number: **None** Page Number(s): **280** ITS Number: **BSI 1b** Bases JFD Number: **None**
Comment	**This is a Beyond Scope Issue (TAC No. MC7598) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your**

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/12/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 02/15/2006	During a phone conversation with the NRC reviewer, the NMC response to whether or not the ATWS-RPT Reactor Vessel Steam Dome Pressure - High Function is an LSSS was discussed. The NRC reviewer requested that the NMC response be revised for clarity. Below is the revised response, and it replaces the previous response provided for Question 2. REVISED RESPONSE Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128. · Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS) · Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS) · Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS) · Reactor Low Water Level Scram (RPS) · Reactor Low Water Level ECCS Initiation (ECCS) · Main Steam Isolation Valve (MSIV) Closure (RPS) · Turbine Control Valve Fast Closure (RPS) · Turbine Stop Valve Closure (RPS) · Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation) CTS Table 3.2.5 Function 1 (Improved Technical Specification (ITS)

	LCO 3.3.4.1.b), Reactor Vessel Steam Dome Pressure - High, is not one of the above LSSS. The Reactor Vessel Steam Dome Pressure - High Function initiates a recirculation pump trip (RPT) for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressure event, the RPT aids in termination of the Anticipated Transient Without Scram (ATWS) event and, along with the safety/relief valves, limits the peak reactor pressure vessel pressure to less than the ASME Section III Code Service Level C Limits (1500 psig). The ATWS-RPT is not assumed to mitigate any accident or transient in the original design or licensing basis in the safety analysis. Therefore, this Function does not meet the criteria for being an LSSS.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   CTS Table 3.2.5 Function 1 (Improved Technical Specification (ITS) LCO 3.3.4.1.b), Reactor Vessel Steam Dome Pressure - High, is not one of the above LSSS.

   The Reactor Vessel Steam Dome Pressure - High Function initiates a recirculation pump trip (RPT) for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressure event, the RPT aids in termination of the Anticipated Transient Without Scram (ATWS) event and, along with the safety/relief valves, limits the peak reactor pressure vessel pressure to less than the ASME Section III Code Service Level C Limits (1500 psig). The ATWS-RPT is not assumed to mitigate any accident or transient in the original design or licensing basis in the safety analysis. The ATWS-RPT System was added to the Monticello design and Technical Specifications to satisfy the final ATWS Rule (10 CFR 50.62). Furthermore, the NRC stated in the Federal Register Notice (Volume 60, No. 138, pages 36954, 36955, and 36956, dated 7/19/95) promulgating final rulemaking concerning 10 CFR 50.36(c)(2), that the ATWS-RPT System met Criterion 4. Criterion 4 is a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Furthermore, the Improved Standard Technical Specifications (ISTS) Bases for the ATWS-RPT Instrumentation (ISTS 3.3.4.1), Applicable Safety Analyses, LCO, and Applicability Sections (Page 299 of 760), also states that the ATWS-RPT Instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). Therefore, this Function does not meet the criteria for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200510281246** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.3** ITS Number: **BSI 1c** DOC Number: **M.8** JFD Number: **None** Bases JFD Number: **None** Page Number(s): **321**
Comment	**This is a Beyond Scope Issue (TAC No. MC7599)** **In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns.** **To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information:** **1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation.** **2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not.** **The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not:** **(a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System.** **(b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS).** **(c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS).** **3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met.** **4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your**

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? No

▼Responses

Licensee Response by Jerry Jones on 12/12/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 04/08/2006	During a phone conversation with the NRC, NMC has agreed to apply the Notes that appear in TSTF-493, Draft 0 on the following ECCS Trip Functions: · CTS Table 3.2.2 Function C.3 (Improved Technical Specification (ITS) Table 3.3.5.1-1 Functions 4.c, 4.d, 5.c, and 5.d), Automatic Depressurization System (ADS) Low Pressure Core Cooling Pumps Discharge Pressure Interlock. In addition, the submittal has been modified to reflect a new Allowable Value for each of these Functions. The Allowable Value changes were made so that all the Allowable Values for these Functions would be the same. The proposed modifications are provided in the attachment to this response. The proposed Notes are consistent with the latest revision of the draft TSTF-493, versus the September 7, 2005 letter from the NRC to NEI. For Note 1, the difference is that the last sentence of the note from the September 7, 2005 letter has been deleted. The sentence in the September 7, 2005 letter states, "If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value the channel shall be declared inoperable." This statement is redundant to the requirements of the instrumentation LCOs, which require the Allowable Values to be met, and LCO

	3.0.1, which dictates the LCOs shall be met during the MODES or other specified conditions. This was the reason that TSTF-493 did not include this sentence. For Note 2, The difference is within the 2nd sentence. The September 7, 2005 letter states The [Limiting Trip Setpoint] and the methodology used to determine the [Limiting Trip Setpoint], the pre-defined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in the UFSAR [or Bases][or a document incorporated into the UFSAR such as the technical requirements manual]. The latest draft of TSTF-493 specifies, "The [Limiting Trip Setting], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band shall be specified in a document controlled under 10 CFR 50.59. This sentence was modified to address two issues. 1) The methodology for determining the [Limiting Trip Setpoint] is proprietary information for many plants, including Monticello. Therefore, placing this information in a document available to the public is not an option. 2) Specifying the 10 CFR 50.59 controlled document within the note would require a license amendment to change the location. Therefore, it was proposed to only specify that the [Limiting Trip Setpoint], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band will be in a document controlled under 10 CFR 50.59. The Bases will indicate the specific document where this information resides, i.e., TRM. This will allow the location of the above information to be changed to another 10 CFR 50.59 document using the 10 CFR 50.59 change process. Additionally, during a later telephone conversation with the NRC, NMC stated the Bases would be modified describing the notes, and would evaluate other Bases information contained in the latest draft revision of TSTF-493 for addition. The results of this evaluation are that we intend to only revise the Bases to the extent necessary to provide a general description of the Notes and include the specific 10 CFR 50.59 controlled document where the methodology is located. Furthermore, due to the addition of the two Notes, ITS Table 3.3.5.1-1 Note (c) has been renumbered to Note (e). The affected ITS pages are also included in the attachment to this response. Current Technical Specification (CTS) Table 3.2.2 Function C.3 "Automatic Depressurization Low Pressure Core Cooling Pumps Discharge Pressure Interlock Trip Setting" had a range of greater than or equal to 60 psig to less than or equal to150 psig (Attachment 1, Volume 8, Rev. 0, Page 321 of 760). The Improved Technical Specifications (ITS) License Amendment Request (LAR) submittal included the following Allowable Values (Pages 356 and 357): Table 3.3.5.1-1 Function 4.c and 5.c: greater than or equal to 65.0 psig to less than or equal to 135.4 psig Table 3.3.5.1-1 Function 4.d and 5.d: greater than or equal to 61.3 psig to less than or equal to 37.7 psig for PS-10-105A through D and greater than or equal to 60.0 psig to less than or equal to 139.8 psig for PS-10-105E through G After submittal of the ITS LAR it was determined that it was preferable to have a common Allowable Value range for both functions as was the case in the CTS. The setpoint calculations were revised to produce a bounding Allowable Value range of greater than or equal to 75 psig and less than or equal to 125 psig applicable to all the switches used for these functions. The use of a common Allowable Value range maintains the simplicity of the CTS while providing an Allowable Value range that conservatively provides for protection of the Analytical Limits. NOTE: The revised ITS submittal pages have been broken up into two attachments. The attachment to this response is part 1 of the changes and the attachment to next response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry	NOTE: The revised ITS submittal pages have been broken up into two

Jones on 04/08/2006	attachments. The attachment to the previous response is part 1 of the changes and the attachment to this response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry Jones on 04/10/2006	Based on a phone conversation between the NRC and NMC, the NRC requested that the in lieu of specifying in ITS Table 3.3.5.1-1 Note (d) that the location for the nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified "in a document controlled under 10 CFR 50.59," that the above information is specified "in the Technical Requirements Manual." Therefore, the ITS submittal changed pages provided as an attachment to the previous NMC response will be changed to annotate that the location is the Technical Requirements Manual.

Date Created: 10/28/2005 12:46 PM by Terry Beltz
Last Modified: 11/29/2005 01:31 PM

2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   CTS Table 3.2.2 Function C.3 (Improved Technical Specification (ITS) Table 3.3.5.1-1 Functions 4.c, 4.d, 5.c, and 5.d), Automatic Depressurization System (ADS) Low Pressure Core Cooling Pumps Discharge Pressure Interlock, is not one of the above LSSS.

   The ADS is designed to provide depressurization of the Reactor Coolant System during a small break Loss of Coolant Accident (LOCA) if the High Pressure Coolant Injection (HPCI) System fails or is unable to maintain required water level in the reactor pressure vessel. The Discharge Pressure Interlock signals from the Core Spray (CS) and Low Pressure Coolant Injection (LPCI) pumps are used as permissives for ADS initiation, indicating that there is a source of low pressure cooling water available once the ADS has depressurized the reactor pressure vessel. Thus, the instruments support the core cooling function of the Emergency Core Cooling System (which are explicitly assumed in the safety analyses), to help ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. However, the 10 CFR 50.46 limits are not directly related to either the fuel cladding integrity minimum critical power ratio safety limit or the reactor coolant pressure boundary safety limit. In addition, these instruments are not specifically credited in the accident analysis. Therefore, this Function does not meet the criteria for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200510281248 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3; ITS Number: BSI 1d; DOC Number: L.5; JFD Number: None; Bases JFD Number: None; Page Number(s): 320
Comment	This is a Beyond Scope Issue (TAC No. MC7600) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 12/12/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
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	pressure ECCS pumps after an ECCS initiation signal is received. However, check valves on the downstream piping (high pressure side) helps to preclude leakage (and thus pressure) to the low pressure side of the CS and LPCI subsystems. Furthermore, both CS and LPCI have relief valves installed on the low pressure side to relieve pressure. In addition, these instruments are not specifically credited in the accident analysis. Therefore, these Functions do not meet the criteria in 10 CFR 50.36 for being an LSSS. b. CTS Table 3.2.8 Function C.1 (ITS Table 3.3.5.1-1 Function 3), High Pressure Coolant Injection (HPCI) System Condensate Storage Tank Level - Low, is not one of the above LSSS. The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. Low level in a Condensate Storage Tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valves between the HPCI pump and the CSTs are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from all aligned CSTs. However, if the water level in any CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. The credited source of water for the HPCI System in the accident analysis is the suppression pool, not the CSTs. Therefore, this Function does not meet the criteria for being an LSSS.
Licensee Response by Jerry Jones on 04/08/2006	During a phone conversation with the NRC, NMC has agreed to apply the Notes that appear in TSTF-493, Draft 0 on the following ECCS Trip Functions: · CTS Table 3.2.2 Function A.1.b.i (Improved Technical Specification (ITS) Table 3.3.5.1?1 Functions 1.d and 2.d), Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Pump Start Reactor Low Pressure Permissive, and · CTS Table 3.2.2 Function A.2 (ITS Table 3.3.5.1-1 Functions 1.c and 2.c), CS and LPCI Low Reactor Pressure (Valve Permissive). The proposed modifications are provided in the attachment to this response. The proposed Notes are consistent with the latest revision of the draft TSTF-493, versus the September 7, 2005 letter from the NRC to NEI. For Note 1, the difference is that the last sentence of the note from the September 7, 2005 letter has been deleted. The sentence in the September 7, 2005 letter states, "If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value the channel shall be declared inoperable." This statement is redundant to the requirements of the instrumentation LCOs, which require the Allowable Values to be met, and LCO 3.0.1, which dictates the LCOs shall be met during the MODES or other specified conditions. This was the reason that TSTF-493 did not include this sentence. For Note 2, The difference is within the 2nd sentence. The September 7, 2005 letter states The [Limiting Trip Setpoint] and the methodology used to determine the [Limiting Trip Setpoint], the pre-defined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in the UFSAR [or Bases][or a document incorporated into the UFSAR such as the technical requirements manual]. The latest draft of TSTF-493 specifies, "The [Limiting Trip Setting], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band shall be specified in a document controlled under 10 CFR 50.59. This sentence was modified to address two issues. 1) The methodology for determining the [Limiting Trip Setpoint] is proprietary information for many plants, including Monticello. Therefore, placing this information in a document available to the public is not an option. 2) Specifying the 10 CFR 50.59 controlled document within the note would require a license amendment to change the location.

	Therefore, it was proposed to only specify that the [Limiting Trip Setpoint], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band will be in a document controlled under 10 CFR 50.59. The Bases will indicate the specific document where this information resides, i.e., TRM. This will allow the location of the above information to be changed to another 10 CFR 50.59 document using the 10 CFR 50.59 change process. Additionally, during a later telephone conversation with the NRC, NMC stated the Bases would be modified describing the notes, and would evaluate other Bases information contained in the latest draft revision of TSTF-493 for addition. The results of this evaluation are that we intend to only revise the Bases to the extent necessary to provide a general description of the Notes and include the specific 10 CFR 50.59 controlled document where the methodology is located. Furthermore, due to the addition of the two Notes, ITS Table 3.3.5.1-1 Note (c) has been renumbered to Note (e). The affected ITS pages are also included in the attachment to this response. NOTE: The revised ITS submittal pages have been broken up into two attachments. The attachment to this response is part 1 of the changes and the attachment to next response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry Jones on 04/08/2006	NOTE: The revised ITS submittal changed pages as a result of the previous response (dated 4/8/06) have been broken up into two attachments. The attachment to the previous response is part 1 of the changes and the attachment to this response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry Jones on 04/10/2006	Based on a phone conversation between the NRC and NMC, the NRC requested that the in lieu of specifying in ITS Table 3.3.5.1-1 Note (d) that the location for the nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified "in a document controlled under 10 CFR 50.59," that the above information is specified "in the Technical Requirements Manual." Therefore, the ITS submittal changed pages provided as an attachment to the previous NMC response will be changed to annotate that the location is the Technical Requirements Manual.

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Last Modified: 11/29/2005 01:32 PM

2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   a. CTS Table 3.2.2 Function A.1.b.i (Improved Technical Specification (ITS) Table 3.3.5.1-1 Functions 1.d and 2.d), Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Pump Start Reactor Low Pressure Permissive, and CTS Table 3.2.2 Function A.2 (ITS Table 3.3.5.1-1 Functions 1.c and 2.c), CS and LPCI Low Reactor Pressure (Valve Permissive), are not one of the above LSSS.

      The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to starting the pumps and opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. Thus, they effectively delay the start of the low pressure ECCS pumps after an ECCS initiation signal is received. However, check valves on the downstream piping (high pressure side) helps to preclude leakage (and thus pressure) to the low pressure side of the CS and LPCI subsystems. Furthermore, both CS and LPCI have relief valves installed on the low pressure side to relieve pressure. In addition, these instruments are not specifically credited in the accident analysis. Therefore, these Functions do not meet the criteria in 10 CFR 50.36 for being an LSSS.

   b. CTS Table 3.2.8 Function C.1 (ITS Table 3.3.5.1-1 Function 3), High Pressure Coolant Injection (HPCI) System Condensate Storage Tank Level - Low, is not one of the above LSSS.

      The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. Low level in a Condensate Storage Tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valves between the HPCI pump and the CSTs are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from all aligned CSTs. However, if the water level in any CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. The Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool. However, the Function is not specifically credited in the accident analysis, and is therefore not considered to be an LSSS as defined in 10 CFR 50.36.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200510281248** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **3.3** DOC Number: **L.3** JFD Number: **None** Page Number(s): **419** ITS Number: **BSI 1e** Bases JFD Number: **None**
Comment	**This is a Beyond Scope Issue (TAC No. MC7601) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your**

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

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	Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS) · Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS) · Reactor Low Water Level Scram (RPS) · Reactor Low Water Level ECCS Initiation (ECCS) · Main Steam Isolation Valve (MSIV) Closure (RPS) · Turbine Control Valve Fast Closure (RPS) · Turbine Stop Valve Closure (RPS) · Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation) CTS Table 3.2.8 Function C.1 (Improved Technical Specification (ITS) Table 3.3.5.2-1, Function 3), Reactor Core Isolation Cooling (RCIC) System Condensate Storage Tank Level - Low, is not one of the above LSSS. The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that injection by the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. Low level in a Condensate Storage Tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the RCIC pump and the CSTs is open and, upon receiving a RCIC initiation signal, water for RCIC injection would be taken from all aligned CSTs. However, if the water level in any CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. No credit is taken in the safety analyses for RCIC System operation, thus the Condensate Storage Tank Level - Low instrumentation is not credited. In addition, the CTS is not credited in the RCIC events as the source of water for the RCIC System; the suppression pool water is the assumed source. Therefore, this Function does not meet the criteria for being an LSSS.
Licensee Response by Jerry Jones on 02/15/2006	During a phone conversation with the NRC reviewer, the NMC response to whether or not the HPCI System Condensate Storage Tank Level - Low Function is an LSSS was discussed. The NRC reviewer requested that the NMC response be revised for clarity. Below is the revised response, and it replaces the previous response provided for Question 2. REVISED RESPONSE Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128. · Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS) · Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS) · Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS) · Reactor Low Water Level Scram (RPS) · Reactor Low Water Level ECCS Initiation (ECCS) · Main Steam Isolation Valve (MSIV) Closure (RPS) · Turbine Control Valve Fast Closure (RPS) · Turbine Stop Valve Closure (RPS) · Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation) a. CTS Table 3.2.2 Function A.1.b.i (Improved Technical Specification (ITS) Table 3.3.5.1-1 Functions 1.d and 2.d), Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Pump Start Reactor Low Pressure Permissive, and CTS Table 3.2.2 Function A.2 (ITS Table 3.3.5.1-1 Functions 1.c and 2.c), CS and LPCI Low Reactor Pressure (Valve Permissive), are not one of the above LSSS. The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. This ensures that, prior to starting the pumps and opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. Thus, they effectively delay the start of the low

	pressure ECCS pumps after an ECCS initiation signal is received. However, check valves on the downstream piping (high pressure side) helps to preclude leakage (and thus pressure) to the low pressure side of the CS and LPCI subsystems. Furthermore, both CS and LPCI have relief valves installed on the low pressure side to relieve pressure. In addition, these instruments are not specifically credited in the accident analysis. Therefore, these Functions do not meet the criteria in 10 CFR 50.36 for being an LSSS. b. CTS Table 3.2.8 Function C.1 (ITS Table 3.3.5.1-1 Function 3), High Pressure Coolant Injection (HPCI) System Condensate Storage Tank Level - Low, is not one of the above LSSS. The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. Low level in a Condensate Storage Tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valves between the HPCI pump and the CSTs are open and, upon receiving a HPCI initiation signal, water for HPCI injection would be taken from all aligned CSTs. However, if the water level in any CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the HPCI pump. The credited source of water for the HPCI System in the accident analysis is the suppression pool, not the CSTs. Therefore, this Function does not meet the criteria for being an LSSS.
Licensee Response by Jerry Jones on 04/08/2006	During a phone conversation with the NRC, NMC has agreed to apply the Notes that appear in TSTF-493, Draft 0 on the following ECCS Trip Functions: · CTS Table 3.2.2 Function A.1.b.i (Improved Technical Specification (ITS) Table 3.3.5.1?1 Functions 1.d and 2.d), Core Spray (CS) and Low Pressure Coolant Injection (LPCI) Pump Start Reactor Low Pressure Permissive, and · CTS Table 3.2.2 Function A.2 (ITS Table 3.3.5.1-1 Functions 1.c and 2.c), CS and LPCI Low Reactor Pressure (Valve Permissive). The proposed modifications are provided in the attachment to this response. The proposed Notes are consistent with the latest revision of the draft TSTF-493, versus the September 7, 2005 letter from the NRC to NEI. For Note 1, the difference is that the last sentence of the note from the September 7, 2005 letter has been deleted. The sentence in the September 7, 2005 letter states, "If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value the channel shall be declared inoperable." This statement is redundant to the requirements of the instrumentation LCOs, which require the Allowable Values to be met, and LCO 3.0.1, which dictates the LCOs shall be met during the MODES or other specified conditions. This was the reason that TSTF-493 did not include this sentence. For Note 2, The difference is within the 2nd sentence. The September 7, 2005 letter states The [Limiting Trip Setpoint] and the methodology used to determine the [Limiting Trip Setpoint], the pre-defined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in the UFSAR [or Bases][or a document incorporated into the UFSAR such as the technical requirements manual]. The latest draft of TSTF-493 specifies, "The [Limiting Trip Setting], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band shall be specified in a document controlled under 10 CFR 50.59. This sentence was modified to address two issues. 1) The methodology for determining the [Limiting Trip Setpoint] is proprietary information for many plants, including Monticello. Therefore, placing this information in a document available to the public is not an option. 2) Specifying the 10 CFR 50.59 controlled document within the note would require a license amendment to change the location.

	Therefore, it was proposed to only specify that the [Limiting Trip Setpoint], the methodology used to determine the as-found acceptance criteria band, and the methodology used to determine the as-left setpoint tolerance band will be in a document controlled under 10 CFR 50.59. The Bases will indicate the specific document where this information resides, i.e., TRM. This will allow the location of the above information to be changed to another 10 CFR 50.59 document using the 10 CFR 50.59 change process. Additionally, during a later telephone conversation with the NRC, NMC stated the Bases would be modified describing the notes, and would evaluate other Bases information contained in the latest draft revision of TSTF-493 for addition. The results of this evaluation are that we intend to only revise the Bases to the extent necessary to provide a general description of the Notes and include the specific 10 CFR 50.59 controlled document where the methodology is located. Furthermore, due to the addition of the two Notes, ITS Table 3.3.5.1-1 Note (c) has been renumbered to Note (e). The affected ITS pages are also included in the attachment to this response. NOTE: The revised ITS submittal pages have been broken up into two attachments. The attachment to this response is part 1 of the changes and the attachment to next response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry Jones on 04/08/2006	NOTE: The revised ITS submittal changed pages as a result of the previous response (dated 4/8/06) have been broken up into two attachments. The attachment to the previous response is part 1 of the changes and the attachment to this response contains part 2 of the changes. It was broken up due to the size of the attachment.
Licensee Response by Jerry Jones on 04/10/2006	Based on a phone conversation between the NRC and NMC, the NRC requested that the in lieu of specifying in ITS Table 3.3.5.1-1 Note (d) that the location for the nominal trip setpoint and the methodology used to determine the as-found tolerance and the as-left tolerance are specified "in a document controlled under 10 CFR 50.59," that the above information is specified "in the Technical Requirements Manual." Therefore, the ITS submittal changed pages provided as an attachment to the previous NMC response will be changed to annotate that the location is the Technical Requirements Manual.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   CTS Table 3.2.8 Function C.1 (Improved Technical Specification (ITS) Table 3.3.5.2-1, Function 3), Reactor Core Isolation Cooling (RCIC) System Condensate Storage Tank Level - Low, is not one of the above LSSS.

   The purpose of the RCIC System instrumentation is to initiate actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink (the main condenser) and normal coolant makeup flow from the Reactor Feedwater System is unavailable, such that injection by the low pressure Emergency Core Cooling Systems (ECCS) pumps does not occur. Low level in a Condensate Storage Tank (CST) indicates the unavailability of an adequate supply of makeup water from this normal source. Normally, the suction valve between the RCIC pump and the CSTs is open and, upon receiving a RCIC initiation signal, water for RCIC injection would be taken from all aligned CSTs. However, if the water level in any CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. This ensures that an adequate supply of makeup water is available to the RCIC pump. No credit is taken in the safety analyses for RCIC System operation, thus the Condensate Storage Tank Level - Low instrumentation is not credited. Furthermore, the NRC stated in the Federal Register Notice (Volume 60, No. 138, pages 36954, 36955, and 36956, dated 7/19/95) promulgating final rulemaking concerning 10 CFR 50.36(c)(2), that the RCIC System met Criterion 4. Criterion 4 is a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Furthermore, the Improved Standard Technical Specifications (ISTS) Bases for the RCIC System Instrumentation (ISTS 3.3.5.1), Applicable Safety Analyses, LCO, and Applicability Sections (Page 440 of 760), also states that the RCIC System Instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). Therefore, this Function does not meet the criteria for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200510281249 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: BSI 1f DOC Number: M.9 JFD Number: None Bases JFD Number: None Page Number(s): 459
Comment	This is a Beyond Scope Issue (TAC No. MC7602) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 12/17/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
Licensee Response by Jerry Jones on 02/15/2006	During a phone conversation with the NRC reviewer, the NMC response to whether or not the HPCI Steam Line isolations (CTS Table 3.2.1 Functions 4.a, 4.b, and 4.c) are LSSS was discussed. The NRC reviewer requested that the NMC response be revised for clarity. Below is the revised response, and it replaces the previous response provided for Question 2. REVISED RESPONSE Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128. · Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS) · Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS) · Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS) · Reactor Low Water Level Scram (RPS) · Reactor Low Water Level ECCS Initiation (ECCS) · Main Steam Isolation Valve (MSIV) Closure (RPS) · Turbine Control Valve Fast Closure (RPS) · Turbine Stop Valve Closure (RPS) · Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation) In general, the primary containment isolation instrumentation

	automatically initiates closure of appropriate primary containment isolation valves (PCIVs), in combination with other accident mitigation systems, to limit fission product release during and following postulated Design Basis Accidents (DBAs). a. CTS Table 3.2.1 Function 3.d (Improved Technical Specification (ITS) Table 3.3.6.1-1 Function 5.a), High Reactor Water Cleanup (RWCU) System Flow Isolation, is not one of the above LSSS. The high flow signal is provided to detect a break in the RWCU System. This will detect leaks in the RWCU System when room temperature would not provide detection (i.e., a cold leg break). Should the reactor coolant continue to flow out of the break, offsite dose limits may be exceeded. Therefore, isolation of the RWCU System is initiated when high flow is sensed to prevent exceeding offsite doses. A time delay is provided to prevent spurious trips during most RWCU operational transients. This Function is not assumed in any Updated Safety Analysis Report (USAR) transient or accident analysis, since bounding analyses are performed for large breaks such as Main Steam Line (MSL) Breaks. Therefore, this Function does not meet the criteria for being an LSSS. b. CTS Table 3.2.1 Function 4.a (ITS Table 3.3.6.1-1 Function 3.a), High Pressure Coolant Injection (HPCI) High Steam Flow Isolation, CTS Table 3.2.1 Function 4.b (ITS Table 3.3.6.1-1 Function 3.c), HPCI Steam Line Area High Temperature Isolation, and CTS Table 3.2.1 Function 4.c (ITS Table 3.3.6.1-1 Function 3.b), Low Pressure in HPCI Steam Supply Line Isolation, are not one of the above LSSS. The HPCI Steam Line Isolation on high flow is initiated to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. HPCI Steam Line Area High Temperature Isolation is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. Specific credit for the high flow and high temperature Functions are not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. The HPCI Steam Supply Line Low Pressure Isolation is for equipment protection and is not assumed in any transient or accident analysis in the USAR. This isolation is included in Technical Specifications because of the potential for risk due to possible failure of the instruments preventing HPCI initiation. Therefore, since these Functions do not affect a Safety Limit they do not meet the criteria for being an LSSS. c. CTS Table 3.2.1 Function 5.b (ITS Table 3.3.6.1-1 Function 4.c), Reactor Core Isolation Cooling (RCIC) Steam Line Area High Temperature Isolation, is not one of the above LSSS. RCIC Steam Line Area High Temperature is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. This Function is not assumed in any USAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks. Therefore, this Function does not meet the criteria for being an LSSS.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   In general, the primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs), in combination with other accident mitigation systems, to limit fission product release during and following postulated Design Basis Accidents (DBAs).

   a. CTS Table 3.2.1 Function 3.d (Improved Technical Specification (ITS) Table 3.3.6.1-1 Function 5.a), High Reactor Water Cleanup (RWCU) System Flow Isolation, is not one of the above LSSS.

      The high flow signal is provided to detect a break in the RWCU System. This will detect leaks in the RWCU System when room temperature would not provide detection (i.e., a cold leg break). Should the reactor coolant continue to flow out of the break, offsite dose limits may be exceeded. Therefore, isolation of the RWCU System is initiated when high flow is sensed to prevent exceeding offsite doses. A time delay is provided to prevent spurious trips during most RWCU operational transients. This Function is not assumed in any Updated Safety Analysis Report (USAR) transient or accident analysis, since bounding analyses are performed for large breaks such as Main Steam Line (MSL) Breaks. Therefore, since this Function is required due to ensuring 10 CFR 100 limits are met (which is not a safety limit), it does not meet the criteria in 10 CFR 50.36 for being an LSSS.

   b. CTS Table 3.2.1 Function 4.a (ITS Table 3.3.6.1-1 Function 3.a), High Pressure Coolant Injection (HPCI) High Steam Flow Isolation, CTS Table 3.2.1 Function 4.b (ITS Table 3.3.6.1-1 Function 3.c), HPCI Steam Line Area High Temperature Isolation, and CTS Table 3.2.1 Function 4.c (ITS Table 3.3.6.1-1 Function 3.b), Low Pressure in HPCI Steam Supply Line Isolation, are not one of the above LSSS.

      The HPCI Steam Line Isolation on high flow is initiated to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. HPCI Steam Line Area High Temperature Isolation is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. Specific credit for the high flow and high temperature Functions are not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and MSL breaks. The HPCI Steam Supply Line Low Pressure Isolation is for equipment protection and is not assumed in any transient or accident analysis in the USAR. This isolation is included in Technical Specifications because of the potential for risk due to possible failure of the instruments preventing HPCI initiation. Therefore, since these Functions are required either due to ensuring 10 CFR 50.46 and 10 CFR 100 limits are met (which are not safety limits) or due to preventing HPCI from operating, they do not meet the criteria in 10 CFR 50.36 for being an LSSS.

c. CTS Table 3.2.1 Function 5.b (ITS Table 3.3.6.1-1 Function 4.c), Reactor Core Isolation Cooling (RCIC) Steam Line Area High Temperature Isolation, is not one of the above LSSS.

   RCIC Steam Line Area High Temperature is provided to detect a leak from the associated system steam piping. The isolation occurs when a very small leak has occurred and is diverse to the high flow instrumentation. If the small leak is allowed to continue without isolation, offsite dose limits may be reached. This Function is not assumed in any USAR transient or accident analysis, since bounding analyses are performed for large breaks such as recirculation or MSL breaks. Therefore, since this Function is required due to ensuring 10 CFR 100 limits are met (which is not a safety limit), it does not meet the criteria in 10 CFR 50.36 for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200510281250 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: BSI 1g DOC Number: L.9 JFD Number: None Bases JFD Number: None Page Number(s): 459
Comment	This is a Beyond Scope Issue (TAC No. MC7603) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, ?Technical Specifications.? The NRC staff has been working with the Nuclear Energy Institute?s Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. **5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements.** **6. Provide commitment to assess applicability of the TSTF?s TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP?s licensing basis are necessary.** **7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated, ? Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift.? Explain what these two verifications are with examples to clarify the significance of these two verifications.**
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 12/17/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.

Date Created: 10/28/2005 12:50 PM by Terry Beltz
Last Modified: 11/29/2005 01:33 PM

2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   In general, the primary containment isolation instrumentation automatically initiates closure of appropriate primary containment isolation valves (PCIVs), in combination with other accident mitigation systems, to limit fission product release during and following postulated Design Basis Accidents (DBAs).

   a. CTS Table 3.2.1 Function 1.b (Improved Technical Specification (ITS) Table 3.3.6.1-1 Function 1.c), High Flow in Main Steam Line Isolation, is not one of the above LSSS.

      High Flow in Main Steam Line Isolation is provided to detect a break of the MSL and to initiate closure of the MSIVs. The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits. However, the 10 CFR 50.46 and 10 CFR 100 limits are not directly related to either the fuel cladding integrity minimum critical power ratio safety limit or the reactor coolant pressure boundary safety limit. Therefore, this Function does not meet the criteria in 10 CFR 50.36 for being an LSSS.

   b. CTS Table 3.2.1 Function 1.d (ITS Table 3.3.6.1-1 Function 1.b), Low pressure in Main Steam Line Isolation is an LSSS at Monticello.

   c. CTS Table 3.2.1 Function 5.a (ITS Table 3.3.6.1-1 Function 4.a), Reactor Core Isolation Cooling (RCIC) High Steam Flow Isolation, and CTS Table 3.2.1 Function 5.c (ITS Table 3.3.6.1-1 Function 4.b), Low Pressure in RCIC Steam Supply Line Isolation, are not one of the above LSSS.

      The RCIC Steam Line Isolation on high flow is initiated to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for the high flow Function is not assumed in any Updated Safety Analysis Report (USAR) accident analyses since the bounding analysis is performed for large breaks such as recirculation and Main Steam Line (MSL) breaks. The RCIC Steam Supply Line Low Pressure Isolation is for equipment protection and is not assumed in any transient or accident analysis in the USAR. This isolation is included in Technical Specifications because of the potential for risk due to possible failure of the instruments preventing RCIC initiation. Therefore, since these Functions are required either due to ensuring 10

CFR 50.46 and 10 CFR 100 limits are met (which are not safety limits) or due to preventing RCIC from operating, they do not meet the criteria in 10 CFR 50.36 for being an LSSS.

d. CTS Table 3.2.1 Function 6.a (ITS Table 3.3.6.1-1 Function 6.a), Shutdown Cooling Supply Isolation Reactor Pressure Interlock, is not one of the above LSSS.

   The Reactor Pressure Interlock is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This interlock is provided only for equipment protection to prevent an intersystem Loss of Coolant Accident scenario, and credit for the interlock is not assumed in the accident or transient analysis in the USAR. Therefore, since this Function is required due to ensuring the RHR System maintains OPERABILITY (which is not a safety limit), it does not meet the criteria in 10 CFR 50.36 for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200510281251 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: BSI 1h DOC Number: M.3 JFD Number: None Bases JFD Number: None Page Number(s): 683
Comment	This is a Beyond Scope Issue (TAC No. MC7604) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, Technical Specifications. The NRC staff has been working with the Nuclear Energy Institute's Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTFs TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGPs licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated that "Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift." Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**10/28/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

▼Responses

Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 12/12/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

- Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
- Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
- Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
- Reactor Low Water Level Scram (RPS)
- Reactor Low Water Level ECCS Initiation (ECCS)
- Main Steam Isolation Valve (MSIV) Closure (RPS)
- Turbine Control Valve Fast Closure (RPS)
- Turbine Stop Valve Closure (RPS)
- Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

CTS Table 3.2.6 Function 1 (Improved Technical Specification (ITS) Table 3.3.8.1-1 Functions 2.a and 2.b), Degraded Voltage Protection, is not one of the above LSSS.

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. Offsite power is the preferred source of power for the 4.16 kV essential buses. The Degraded Voltage instrumentation monitors the 4.16 kV essential buses. A reduced voltage condition on a 4.16 kV essential bus indicates that, while offsite power may not be completely lost to the respective essential bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the bus is transferred from offsite power to onsite emergency diesel generator power when the voltage on the bus drops below the setpoint for the applicable amount of time. This ensures that adequate power will be available to the required equipment.

Thus, the instruments support the core cooling function of the Emergency Core Cooling System (which are explicitly assumed in the safety analyses), to help ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. However, the 10 CFR 50.46 limits are not directly related to either the fuel cladding integrity minimum critical power ratio safety limit or the reactor coolant pressure boundary safety limit. In addition, the design basis Loss of Coolant Accident safety analysis does not assume a degraded voltage condition, but a complete loss of offsite power condition. CTS contains requirements to monitor for a complete loss of voltage condition (CTS Table 3.2.6 Function 2, Loss of Voltage Protection. Furthermore, the degraded voltage requirements were added to the Monticello CTS as part of License Amendment 3, dated 3/27/81. These requirements were added as a result of the NRC generic letter issued to Northern States Power, dated 6/3/77. Inclusion of this degraded voltage instrumentation did not include classifying the instrumentation as an LSSS. Therefore, this Function does not meet the criteria for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200511031435 Conference Call Requested? Yes
Category	**Major Technical**
ITS Information	**ITS Section:** 3.6 **ITS Number:** 3.6.1.3 **DOC Number:** LA.1 **JFD Number:** None **Bases JFD Number:** None **Page Number(s):** 66
Comment	**What was done on Cook is an error in my opinion. This convention for using LA-type for R-type changes in which the LCO meets none of the criteria is contrary to past conversions, and plant-specific relocations approved apart from a conversion amendment. Regarding snubbers, why has Monticello chosen to not adopt TSTF-372 - effectively making an R-type change into an A-type change. General comment regarding description of LA-type change Category 6, "Removal of LCO, SR, or other TS requirement to the TRM, USAR, ODCM, OQAP, IST Program, or IIP." It is possible to read "LCO requirement" as meaning "LCO,' instead of just a detail from a CTS LCO. Of course, removal of an [entire specified] LCO requires an R-type designation and a justification for why it meets none of the LCO criteria of 50.36, or if it does, why it is not necessary to be in TSs. So, revise the description to avoid confusion. Closeout comment: Based on adoption of TSTF-372, which adds a general LCO for snubbers, the concerns of this comment are mute. Therefore, this comment is closed.**
Issue Date	11/03/2005

Close Date	02/24/2006 Resolution requires change to: None Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 01/20/2006	This response is provided to address the first paragraph of the NRC comment. The first paragraph of the NRC comment was added after Monticello responded to the original NRC comment, which only included the words starting with "General comment regarding..." The NRC first paragraph is the response to the original Monticello response. The use of an "LA" type Discussion of Change (DOC) to relocate an entire LCO has been done during many ITS conversions. For example, the snubber LCO has been relocated using an "LA" type DOC in lieu of an "R" type DOC during the ITS conversions of DC Cook Units 1 and 2 (License Amendments 287 and 269, dated 6/1/05), Quad Cities Units 1 and 2 (License Amendments 199 and 195, dated 3/30/01), Dresden Units 2 and 3 (License Amendments 185 and 180, dated 3/30/01), LaSalle Units 1 and 2 (License Amendments 147 and 133, dated 3/30/01), Nine Mile Point Unit 2

	(License Amendment 91, dated 2/15/00), and Brunswick Units 1 and 2 (License Amendments 203 and 233, dated 6/5/88). Furthermore, Monticello cannot state that the snubber requirements do not meet any of the criteria specified in 10 CFR 50.36(c)(2)(ii), thus it cannot be classified as an "R" type DOC. In addition, Monticello will be adopting TSTF-372. See the second Monticello response to RAI 200512151125.
Licensee Response by Jerry Jones on 11/08/2005	Improved Technical Specification (ITS) 3.6.1.3 Discussion of Change (DOC) LA.1 (Attachment 1, Volume 11, Rev. 0, Page 66 of 431) is a Type 6 discussion. A Type 6 designation covers the removal of LCO, SR, or other Technical Specification requirements to the Technical Requirements Manual (TRM), Updated Safety Analysis Report (USAR), Offsite Dose Calculation Manual (ODCM), Operational Quality Assurance Program (OQAP), Inservice Testing (IST) Program or Inservice Inspection Program (IIP). The NRC reviewer's comment is concerned that this type of DOC may be used to relocate an entire LCO instead of a detail of an LCO. The comment also states that an entire LCO must be relocated using an R type DOC designation and the R DOC justification must include why it meets none of the criteria of 10 CFR 50.36(c)(2)(ii). An R type DOC is used only if the LCO is being relocated because it does not meet any of the criteria of 10 CFR 50.36(c)(2)(ii). It cannot be used if the LCO does meet one of the criteria. In the case where an LCO being relocated meets at least one of the criteria of 10 CFR 50.36(c)(2)(ii), the Monticello ITS submittal has used the LA Type 6 designation to relocate the entire LCO without an evaluation with respect to the four criteria associated with 10 CFR 50.36(c)(2)(ii). Specifically, an LA Type 6 DOC (Attachment 1, Volume 9, Rev. 0, Page 239 of 255) has been used to relocate the snubber requirements of CTS 3/4.6.H (Pages 234 through 238 of 255) and an LA Type 6 DOC (Attachment 1, Volume 14, Rev. 0, Page 146 of 157) has been used to relocate the decay time requirements of CTS 3.10.D (Page 145 of 157). These DOCs do not provide any discussion concerning the four criteria of 10 CFR 50.36(c)(2)(ii). This type of DOC (LA Type 6) has been used since the two requirements (snubbers and decay time) meet one of the criteria of 10 CFR 50.36(c)(2)(ii), but are not included in the Improved Standard Technical Specifications (ISTS). Thus, they need to be relocated from the Monticello Current Technical Specifications, but cannot be relocated using an R type DOC. This approach (use of an LA type 6 DOC) was also used in the recently approved DC Cook ITS submittal. Therefore, there is no need to revise the LA Type 6 designation.

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NRC ITS TRACKING

NRC Reviewer

ID	**200511071932** Conference Call Requested? **Yes**
Category	**Minor Technical**
ITS Information	ITS Section: **3.6** ITS Number: **3.6.1.3** DOC Number: **None** JFD Number: **12** Bases JFD Number: **None** Page Number(s): **82**
Comment	**Point out the difference between the frequencies of STS SR 3.6.1.3.7 and corresponding ITS SR 3.6.1.3.11. In other words, what does "In accordance with the PCLRTP" require vs. "184 days" (6 months)?**
Issue Date	**11/07/2005**

Close Date	**01/17/2006** Resolution requires change to: NUREG Bases Markup Docket Response Required? **Yes**

Responses

Licensee Response by Jerry Jones on 11/14/2005	Improved Standard Technical Specification (ISTS) SR 3.6.1.3.7 (Attachment 1, Volume 11, Rev. 0, Page 82 of 431) requires the performance of leakage rate testing for each primary containment purge valve with resilient seals every "184 days" and "Once within 92 days after opening the valve." In Monticello Improved Technical Specification (ITS) SR 3.6.1.3.11 (Page 62 of 431), the Surveillance Frequency is "In accordance with the Primary Containment Leakage Rate Testing Program." This Frequency is consistent with Current Technical Specification (CTS) 4.7.D.4 (Page 56 of 431), which states, in part, "If periodic Type C leakage testing of the valves identifies a common mode test failure attributable to seat seal degradation, then the seat seals of all drywell and suppression chamber 18-inch purge and vent valves shall be replaced." Discussion of Change (DOC) A.11 (Page 82 of 431) clarifies that the periodic Type C leakage testing of the drywell and suppression chamber 18-inch purge and vent valves is in accordance with the Primary Containment Leakage Testing Program, which requires type C testing of these valves. Furthermore, the Monticello CTS does not require leakage rate testing of the drywell and suppression chamber 18-inch purge and vent valves more frequently than the normal Type C testing Frequency. The requirements for the "Primary Containment Leakage Rate Testing Program" are located in ITS 5.5.11 (Attachment 1, Volume 17, Rev. 0, Pages 92, 93, 94, and 95 of 143), which requires a program to establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Bases Containment Leak-Test Program," dated September, 1995." 10 CFR 50,

	Appendix J, Option B, Section III.B, states that the Type C leakage rate test must be conducted (1) prior to initial criticality, and (2) periodically thereafter at intervals based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release to reduce the risk from reactor accidents. Furthermore, Regulatory Guide 1.163, Regulatory Position C.2 states that the interval for Type C tests for containment purge and vent valves should be limited to 30 months. Therefore, the drywell and suppression chamber 18-inch purge and vent valve testing Frequency is ineligible for performance-based extended test intervals and the Type C leak test for these valves is currently being performed at an interval not to exceed 30 months. In addition, it was noted that the first paragraph of the first paragraph of the Bases for ITS SR 3.6.1.3.11 (Page 104 of 431), the insert description of the Frequency is incorrect. The currently stated words of "The Frequency of this SR is in accordance withthe Primary" should be "The Frequency of this SR is in accordance with the Primary Containment Leakage Rate Testing Program." This will be corrected as shown in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	**200511072109** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	**ITS Section:** 3.6 **ITS Number:** 3.6.1.3 **DOC Number:** A.4 **JFD Number:** None **Bases JFD Number:** None **Page Number(s):** 59
Comment	**DOC A4 states, "This change is acceptable because the addition of the Note reflects the CTS allowance to take the appropriate Actions on a per valve basis." Explain how Monticello finds justification in the CTSs for timing action requirements 3.7.D.2.a and 3.7.D.2.b on a per valve basis (which is less restrictive than on a "per penetration" basis), when it appears from the CTS language for these actions that the clock is on a condition basis, in which case the addition of ACTIONS Note 2 would be less restrictive, hence an L-type change, Category 4.**
Issue Date	**11/07/2005**

Close Date	**01/17/2006** Resolution requires change to: DOC Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 11/11/2005	Current Technical Specification (CTS) 3.7.D.2.a (Attachment 1, Volume 11, Rev. 0, Page 55 of 431) states, in part, in the event one or more penetration flow paths with one PCIV inoperable, reactor operation may continue "provided that within the subsequent 4 hours (8 hours for MSIVs and 72 hours for EFCVs)" at least one valve in each line having an inoperable valve is deactivated in the isolated condition. CTS 3.7.D.2.b (Page 55 of 431) states, in part, in the event one or more penetration flow paths with two PCIVs inoperable, reactor operation may continue "provided that within the subsequent 1 hour" at least one valve in each line having inoperable valves is deactivated in the isolated condition. These requirements are incorporated into Improved Technical Specification (ITS) 3.6.1.3 ACTIONS A, B, and C (Pages 74 through 76 of 431). Furthermore, ITS 3.6.1.3 ACTIONS Note 2 (Page 74 of 431) states that Separate Condition entry is allowed for each penetration flow path. The Note was added as described in Discussion of Change (DOC) A.4 (Page 59 of 431). The addition of the Note is administrative because the Note reflects the Monticello CTS allowance to take the appropriate Actions on a per penetration flow path basis. (Note: the wording in DOC A.4 is in error when describing the CTS allowance to take Actions on a per "valve" basis, and should have stated on a per "penetration flow path" basis. See the last paragraph of the Monticello response for further discussion.) The proposed 3.6.1.3 ACTIONS A, B, and C and Note 2 are also consistent with the Improved Standard Technical Specification (ISTS) 3.6.1.3. The Monticello CTS

	does not include explicit rules on how to apply the Actions (e.g., on a "condition" basis or on a "component" basis), unlike the ISTS, which does include specific rules. However, the wording of CTS 3.7.D.2.a and 3.7.D.2.b is intended to allow entry into the two Actions on a penetration flow path basis. In addition, the CTS Bases for the two Actions further clarifies the intent of CTS 3.7.D.2.a and 3.7.D.2.b, and states "With one or more penetration flow paths with one PCIV inoperable, the affected penetration must be returned to operable status or isolated within 4 hours (8 hours for MSIVs and 72 hours for Excess Flow Check Valves (EFCVs)). With one or more penetrations with two PCIVs inoperable, either the inoperable PCIVs must be returned to operable status or the affected penetration flow path must be isolated within 1 hour." The CTS Bases words specifically use the singular terms "the affected penetration" and the "affected penetration flow path" when describing the actions that must be taken when one or both PCIVs in one or more penetration flow paths are inoperable. The CTS Bases also states, when describing the justification for the 4 hour completion time, that the "4 hour completion time is reasonable considering the time required to isolate the penetration." The justification for the 4 hour completion time is also clearly based on a penetration flow path basis (considering the time required to isolate "the" penetration). CTS 3.7.D.2.a and CTS 3.7.D.2.b were added to the Technical Specifications and approved in License Amendment 130. Section 3.3 of the NRC Safety Evaluation related to License Amendment 130 discusses the addition of these Actions, and states "The NRC staff has reviewed the proposed changes to CTS 3/4.7.D and finds them acceptable, based upon the above and consistent with NUREG-1433." NUREG-1433 (the BWR/4 ISTS) clearly considers the application to be on a "penetration flow path" basis. Therefore, the Monticello CTS Actions do include the allowance presented in the ITS 3.6.1.3 ACTIONS Note - the Actions are to be taken on a penetration flow path basis. Thus, the change to the Monticello CTS is an administrative change. However, it was noted that the first sentence of the second paragraph of DOC A.4 states "the CTS allowance to take the appropriate Actions on a per valve basis." This is incorrect and should have stated "the CTS allowance to take the appropriate Action on a per penetration flow path basis." This will be corrected as shown in the attachment to this response.

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NRC ITS TRACKING

NRC Reviewer

ID	**200511161501** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	**ITS Section:** 5.0 **ITS Number:** 5.5 **DOC Number:** None **JFD Number:** 16 **Bases JFD Number:** None **Page Number(s):** 87
Comment	**JFD 16 The sentence that you propose to delete contains references to documents that the staff determined were necessary to provde the details for correctly determining the gaseous radioactivity quantities and liquid radwaste quantities. These references are not redundant to ITS 5.5.7 parts a, b, and c. Provide assurance that the sentece with the references will not be deleted.**
Issue Date	**11/16/2005**

Close Date	**02/13/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

NRC Response by Pete Hearn on 01/18/2006	The sentence you propose to delete provides the acceptable methodology for calculating the gaseous radioactive quantities in the Explosive Gas and Storage Tank Radioactivity Monitoring Program. Section 5.5.7 a, b and c do not contain or reference this methhodology information. This methodology information information is required to be in the Radioactivity Monoriting Program. Justify deleting the methodology information.
Licensee Response by Jerry Jones on 01/11/2006	Improved Standard Technical Specification (ISTS) ISTS 5.5.9 (Attachment 1, Volume 17, Rev. 0, Page 87 of 143) includes the following requirements: "The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"]." Improved Technical Specification (ITS) 5.5.7 does not include these requirements and were deleted as justified in Justification for Deviation (JFD) 16 (Pages 98 and 99 of 143), which states, "The program details of the Explosive Gas and Storage Tank Radioactivity Monitoring Program are described in ISTS 5.5.9 (ITS 5.5.7) parts a, b, and c. Therefore, the sentence in the introductory paragraph that specifies a method to determine the explosive gas and storage tank radioactivity is not necessary." Current Technical Specification (CTS) 6.8.I (Page 58 of 143) specifies the requirements for the Explosive Gas and Storage Tank Radioactivity Monitoring

	Program. CTS 6.8.I does not include the statements concerning the methodology for determining the gaseous radioactivity quantities and the liquid radwaste quantities. However, CTS 6.8.I includes the actual values of the quantity of radioactivity in the gas storage tank and temporary outside tank. These values were approved in License Amendment 120, dated 7/24/01. The methodologies were not required to be included in CTS 6.8.I. As part of the conversion to the ITS, the CTS 6.8.I gaseous radioactivity and the liquid radwaste quantity limits have been relocated to the Technical Requirements Manual (TRM) as justified in Discussion of Change (DOC) LA.2 (Page 73 of 143). This is acceptable because ITS 5.5.7 still retains the requirement to include a program that provides controls for potentially explosive gas mixtures contained in the Offgas Treatment System and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. Also, relocating the actual limits to the TRM and not including the specific methodology is consistent with a recently approved ITS conversion, DC Cooks Units 1 and 2 (License Amendments 287 and 269, respectively), dated 6/1/05.
Licensee Response by Jerry Jones on 02/10/2006	After further discussion with the NRC, Monticello will add back into ITS 5.5 the specific values for the two limits in question, consistent with the current licensing basis values. The proposed ITS submittal changes are included in the attachment to this response.

Date Created: 11/16/2005 03:01 PM by Pete Hearn
Last Modified: 02/13/2006 11:08 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200511161520**		Conference Call Requested? **No**	
Category	**Major Technical**			
ITS Information	ITS Section: **5.0** ITS Number: **5.5**	DOC Number: **None**	JFD Number: **14** Bases JFD Number: **None**	Page Number(s): **89**
Comment	**ITS 5.5.8 The proposed term "within limits" for fuel oil testing is undefined. Provide references or specifications for the limits of the fuel oil characteristics required in ITS 5.5.8.**			
Issue Date	**11/16/2005**			

Close Date	**01/12/2006**	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 11/18/2005	Improved Standard Technical Specification (ISTS) 5.5.10 (Attachment 1, Volume 17, Rev. 0, Pages 89 and 90 of 143) provides the requirements for the Diesel Fuel Oil Testing Program. The first paragraph of ISTS states: "A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil to be established. The program shall include sampling and testing requirements, and acceptable criteria, all in accordance with applicable ASTM Standards." It also states the purpose of the program is to establish acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has: an API gravity or an absolute specific gravity within limits, a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and a clear and bright appearance with proper color or a water and sediment content within limits. Furthermore, it requires within 31 days following addition of the new fuel to the storage tanks, other properties for new fuel (other than those listed above) to be within limits. The ISTS does not define the term "within limits" in ISTS 5.5.10. The Monticello Improved Technical Specification (ITS) 5.5.8 (Pages 89 and 90 of 143) maintains the use of the term "within limits." The limits for the ISTS 5.5.10 (ITS 5.5.8) fuel oil testing requirements are specified in the ISTS SR 3.8.3.3 (ITS SR 3.8.3.3) Bases (Attachment 1, Volume 13, Rev. 0, Pages 133 and 134 of 294). The ISTS SR 3.8.3.3 Bases states "The tests, limits, and applicable ASTM Standards are as follows: a. Sample the new fuel oil in accordance with ASTM D4057-[ ] (Ref. 6); b. Verify in accordance with the tests specified in ASTM D975-[ ] (Ref. 6) that the sample has an absolute specific gravity at 60/60¡ãF of ¡Ý 0.83 and ¡Ü 0.89 or an API gravity at 60¡ãF of ¡Ý 27¡ã and ¡Ü 39¡ã when tested in accordance with ASTM D1298-[ ] (Ref. 6), a kinematic viscosity at 40¡ãC of ¡Ý 1.9 centistokes

	and ¡Ü 4.1 centistokes, and a flash point of ¡Ý 125¡ãF, and c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-[ ] or a water and sediment content within limits when tested in accordance with [ASTM D2709-[ ]] (Ref. 6)." Furthermore, the Bases states "Within [31] days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-[ ] (Ref. 6) are met for new fuel oil when tested in accordance with ASTM D975-[ ] (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-[ ], ASTM D2622-[ ], or ASTM D4294-[ ] (Ref. 6)." The Monticello ITS SR 3.8.3.3 Bases has maintained similar information, modified to reflect changes made to ISTS 5.5.10 and to reflect the correct ASTM standards and appropriate limits in use at Monticello.

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NRC ITS TRACKING

NRC Reviewer

ID	200511171133	Conference Call Requested? No
Category	Editorial	
ITS Information	ITS Section: 5.0 ITS Number: 5.6 DOC Number: None	JFD Number: 4 Bases JFD Number: None Page Number(s): 126
Comment	JFD 4 ITS 3.4.3 should be ITS 3.4.9.	
Issue Date	11/17/2005	

Close Date	01/12/2006	Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 11/18/2005	Justification for Deviation (JFD) 4 will be corrected as shown in the attachment to this response.

Date Created: 11/17/2005 11:33 AM by Pete Hearn
Last Modified: 01/12/2006 03:25 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200511171905 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.6 ITS Number: 3.6.1.3 DOC Number: None JFD Number: 6 Bases JFD Number: None Page Number(s): 77
Comment	Add the phrase "for isolation devices outside primary containment" should to the first CT for RA C.2 for consistency with the first CT for RA A.2
Issue Date	11/17/2005

Close Date	01/17/2006	Resolution requires change to: NUREG Markup NUREG Bases Markup Typed ITS Bases None JFD Typed ITS Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 12/01/2005	The first Completion Time in Improved Technical Specification (ITS) 3.6.1.3 Required Action C.2 (Page 77 of 431) will be changed to ?Once per 31 days for isolation devices outside primary containment" as shown in the attachment to this response. Furthermore, Justification for Deviation (JFD) 6 (Page 85 of 431) will be modified to justify the change to first Completion Time of ITS 3.6.1.3 Required Action C.2 and the Bases for the Required Action (Page 95 of 431 will be modified to reflect the changed Completion Time. These changes are also shown in the attachment to this response. In addition, during the development of the Monticello response to this NRC question, it was noted that the Required Action C.2 Bases (Page 95 of 431) was not corrected to incorporate the addition of the Required Action C.2 second Completion Time. Therefore, the Required Action C.2 Bases will be modified to be consistent with the Required Action A.2 Bases discussion of the second Completion Time. This change is shown in the attachment to this response.

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Close

NRC ITS TRACKING

NRC Reviewer

ID	200511282005 Conference Call Requested? No
Category	Major Technical
ITS Information	ITS Section: 3.6 ITS Number: 3.6.1.3 DOC Number: L.6 JFD Number: None Bases JFD Number: None Page Number(s): 89
Comment	CTS 3.7.D.3.a specifies that inerting and de-inerting operations permitted by TS 3.7.A.5.b shall be via the 18 inch purge and vent valves aligned to the Reactor Building plenum and vent, and that all other purging and venting, when primary containment integrity is required, shall be via the 2 inch purge and vent valve bypass line and the Standby Gas Treatment Systems. The CTS Bases for TS 3.7.D and the ITS Bases (application page 89 indicate that the above CTS requirement to use the 2 inch purge and vent bypass will prevent high pressure from reaching the SGT System filter trains in the unlikely event of a loss of coolant accident (LOCA) during venting. Clarify the justification in DOC L.6 for allowing the 18-inch valves to be opened during Modes 1, 2, and 3 for purging and venting, as long as they are opened under approved administrative controls. That is, whenever would the 2-inch valves be insufficient for venting (or for any of the other reasons stated in the proposed ITS Note) in Modes 1, 2, and 3? Also, what is the required time for the 18-inch valves to close under accident conditions, and what is the basis for that time?
Issue Date	11/25/2005

Close Date	01/17/2006 Resolution requires change to: DOC Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 12/09/2005	Current Technical Specification (CTS) 3.7.D.3.a (Attachment 1, Volume 11, Rev. 0, Page 56 of 431) specifies that inerting and de-inerting operations permitted by TS 3.7.A.5.b shall be via the 18 inch purge and vent valves aligned to the Reactor Building plenum and vent and that all other purging and venting, when primary containment integrity is required, shall be via the 2 inch purge and vent valve bypass line and the Standby Gas Treatment System. Discussion of Change (DOC) L.6 (Pages 70 and 71 of 431) provides the allowance to allow the 18 inch purge and vent valves to be opened under more conditions. Improved Standard Technical Specification (ISTS) SR 3.6.1.3.2 (Improved Technical Specification (ITS) SR 3.6.1.3.1) (Page 81 of 431) allows the 18 inch purge and vent valves to be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. The CTS requirement that the 18 inch valves be aligned to the Reactor

	Building plenum and vent has been relocated to the ITS SR 3.6.1.3.1 Bases. This continues to ensure that when the 18 inch valves are open, they are not aligned to the Standby Gas Treatment System. The 2 inch valves are currently used for all pressure control and ALARA or air quality considerations for personnel entry, and are normally acceptable for these purposes. The new allowances for opening the 18 inch valves (pressure control, ALARA or air quality considerations for personnel entry, and Surveillances that require the valves to be open) were added to the Monticello ITS to be consistent with the ISTS. During the development of the ISTS, these new allowances (to allow the large purge valves to be opened for specific reasons) were added in place of the previous time limit requirement (in the previous BWR Standard Technical Specifications, the large purge valves were only allowed to be opened for a maximum amount of time during the past 365 days). Some reasons the large valves might need to be opened would be to minimize the time required to vent the drywell for ALARA or air quality considerations (for personnel safety) and to vent the drywell at a faster rate to preclude a drywell pressure scram (when pressure is increasing in the drywell due to loss of drywell cooling). ITS 3.6.1.3 DOC L.6 justifies the change, stating "use of the purge and vent valves will continue to be minimized and limited to safety related reasons. In addition, these valves are fully qualified to close in the required time under accident conditions to isolate the affected penetrations." USAR Table 5.2-3b, the Primary Containment Automatic Isolation Valves Table, specifies that the 18 inch purge and vent valves are required to close in 15 seconds. This ensures, as stated in the NRC Safety Evaluation Report for Amendment 64 (letter from J. J. Stefano (NRC) to D. M. Musolf (NSP), dated 5/3/89) that the purge supply and vent valves would be closed before the onset of fuel failure following a loss of coolant accident. This is stated in the ITS 3.6.1.3 Applicable Safety Analyses Section of the Bases, Page 90 of 431 (INSERT 4).

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NRC ITS TRACKING

NRC Reviewer

ID	200511291342 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: BSI 9 DOC Number: L.6 JFD Number: None Bases JFD Number: None Page Number(s): 167
Comment	This is a Beyond Scope Issue (TAC No. MC8887) In recent public communications available on the NRC's public website in the Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML052500004, ML050870008 and ML051660447, the Nuclear Regulatory Commission (NRC) staff has identified a concern on using Allowable Values (AV) as limits in Technical Specifications (TS) to satisfy the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, Technical Specifications. The NRC staff has been working with the Nuclear Energy Institute's Setpoint Methods Task Force (TSTF) to revise the TSs to address these concerns. To assess the acceptability of your license amendment request related to this issue, the NRC staff requests the following additional information: 1. Describe the instrumentation setpoint methodology used at Monticello Nuclear Generating Plant (MNGP) for establishing TS limits. This discussion should include acceptable as-found band, acceptable as-left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation. 2. For the setpoint to be revised, clarify whether it is a Limiting Safety System Setting (LSSS) as discussed in 10 CFR 50.36(c)(ii)(A). If you determined that it is not, explain why not. The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of this LSSS issue or not: (a) Instrument setpoints for TS functions in the Reactor Protection (Trip) System. (b) Instrument setpoints for TS functions that protect a safety limit (whether or not the Bases designates the function as an LSSS). (c) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS). 3. 10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the plant setpoint methodology are used to establish the operability of the safety system. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other than the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met. 4. 10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the plant setpoint methodology will ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your

	setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy the requirements of 10 CFR 50.36. 5. For setpoints that are not defined as LSSS in response to Question 2, discuss what measures have been taken to ensure that it is capable of performing its specified safety functions. Include in your discussion information on the controls you employ to ensure that the as-left trip setting after completing periodic surveillances is consistent with your setpoint methodology. If the controls are located in a document other than the TS (e.g., plant test procedure), discuss how those controls satisfy operability requirements. 6. Provide commitment to assess applicability of the TSTFs TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGPs licensing basis are necessary. 7. In discussion of the changes (DOCs) for AVs, e.g. L.12 for ITS 3.3.1.1, it is stated that "Two separate verifications are performed for the calculated NTSP. The first, a Spurious Trip Avoidance Test, evaluates the impact of the NTSP on plant availability. The second verification, an LER Avoidance Test, calculates the probability of avoiding a Licensee Event Report (or exceeding the Allowable Value) due to instrument drift." Explain what these two verifications are with examples to clarify the significance of these two verifications.
Issue Date	**11/29/2005**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/12/2005	The Monticello response to Question 2 is provided in the attachment to this response. The Monticello response to Questions 1 and 3 through 7 has already been provided.
Licensee Response by Jerry Jones on 12/07/2005	The Monticello response to Questions 1 and 3 through 7 is provided in the attachment to this response. The Monticello response to Question 2 will be provided at a later date.
Licensee Response by Jerry Jones on 03/30/2006	During a phone conversation with the NRC reviewer, the NMC response to whether or not the RBM downscale Function is an LSSS was discussed. The NRC reviewer requested that the NMC response be revised for clarity. Below is the revised response and it replaces the previous response provided for Question 2. REVISED RESPONSE Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128. • Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS) • Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS) • Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS) • Reactor Low Water Level Scram (RPS) • Reactor Low Water Level ECCS Initiation (ECCS) • Main Steam Isolation Valve (MSIV) Closure (RPS) • Turbine Control Valve Fast Closure (RPS) • Turbine Stop Valve Closure (RPS) • Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation) CTS Table 3.2.3 Function 4.b (Improved Technical Specification (ITS) Table 3.3.2.1-1 Function

	1.e), Rod Block Monitor (RBM) Downscale, is not one of the above LSSS. The purpose of CTS Table 3.2.3 Function 4.b, RBM Downscale (ITS Table 3.3.2.1-1 Function 1.e) is to preclude rod movement with an inoperable RBM. The RBM Downscale Function is not assumed in any transient or accident analysis in the Updated Safety Analysis Report (USAR). It is provided as an aid to the operators to ensure the RBM is OPERABLE when needed. Normal plant procedures are sufficient to ensure the necessary channels of RBM are OPERABLE prior to rod movement. Therefore, this Function does not meet the criteria for being an LSSS.

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2. Below are the instrumentation setpoints that were previously listed in the Limiting Safety System Settings (LSSS) Table in the Current Technical Specifications (CTS), but were incorporated into Section 3 of the Monticello CTS in Amendment 128.

   - Neutron Flux Intermediate Range Monitor (IRM) - High-High (RPS)
   - Flow Referenced Neutron Flux Average Power Range Monitor (APRM) - High-High (RPS)
   - Flow Referenced Neutron Flux APRM - High Flow Clamp (RPS)
   - Reactor Low Water Level Scram (RPS)
   - Reactor Low Water Level ECCS Initiation (ECCS)
   - Main Steam Isolation Valve (MSIV) Closure (RPS)
   - Turbine Control Valve Fast Closure (RPS)
   - Turbine Stop Valve Closure (RPS)
   - Main Steam Line Low Pressure Initiates MSIV Closure (Primary Containment Isolation)

   CTS Table 3.2.3 Function 4.b (Improved Technical Specification (ITS) Table 3.3.2.1-1 Function 1.e), Rod Block Monitor (RBM) Downscale, is not one of the above LSSS.

   The RBM is designed as an operational aid to assist the reactor operator by initiating a rod block to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. This is protected by CTS Table 3.2.3 Function 4.a (ITS Table 3.3.2.1-1 Functions 1.a, 1.b, and 1.c). However, the purpose of the CTS Table 3.2.3 Function 4.b (ITS Table 3.3.2.1-1 Function 1.e) is to preclude rod movement with an inoperable RBM. In addition, the RBM is designed to allow one RBM channel to be bypassed, and this bypass design is allowed to be used for up to 24 hours in the CTS and ITS. The RBM Downscale Function is not assumed in any transient or accident analysis in the Updated Safety Analysis Report (USAR). It is provided as an aid to the operators to ensure the RBM is OPERABLE when needed. Normal plant procedures are sufficient to ensure the necessary channels of RBM are OPERABLE prior to rod movement. Therefore, this Function does not meet the criteria for being an LSSS.

1. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, Requests for Additional Information (RAI) were sent to Monticello that asked for information similar to information requested in Question 1 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738 and Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello responses were provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1 and in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, these previous Monticello responses, with the exception of the first and last paragraphs of the second listed response, are applicable to Question 1 of this Open Item.

3. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 3 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.b) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.b. Therefore, this previous Monticello response is applicable to Question 3 of this Open Item.

4. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello that asked for information similar to information requested in Question 4 of this Open Item (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Third Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated June 3, 2005 (See NRC RAI 1.a) -- Reference ADAMS ML051230157). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), Enclosure 1, NRC Request #1.a. Therefore, this previous Monticello response, with the exception of the first and last paragraphs, is applicable to Question 4 of this Open Item. Furthermore, in the Monticello response letter identified above, Monticello made the following commitment: "NMC commits to continue resetting Limiting Safety System

Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints has been approved by the NRC and assessed for applicability to MNGP." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

5. Monticello procedures do not differentiate between LSSS and non-LSSS Technical Specification setpoints. Therefore, except for the determination of the Analytical Limit (AL), the discussion presented in the answer to Question 4 applies to this question.

   ALs for those setpoints that are not considered to be an LSSS as defined in 10 CFR 50.36 may be determined from a number of sources. The AL may represent the assumed value input to plant analysis, a design limit derived from industry codes and standards, or engineering judgment when no obvious limit applies to the setpoint.

   Once an AL is determined, the relationship between the AL, Allowable Value, and trip setpoint are the same as discussed in the answer to Question 4.

6. Monticello has previously agreed to provide this commitment. In the letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated July 1, 2005 (Reference ADAMS ML051890051), NMC made the following commitment: "NMC commits to assess applicability of the Technical Specification Task Force's TS changes pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary." Acceptance of this commitment is contained in the letter from NRC to NMC "Monticello Nuclear Generating Plant - Issuance of Amendment Re: Implementation of 24-Month Fuel Cycles (TAC No. MC3692)," dated September 30, 2005 (Reference ADAMS ML052700252).

7. During the NRC review of the Monticello License Amendment Request to revise the Monticello Technical Specifications to support 24-month fuel cycles, a Request for Additional Information (RAI) was sent to Monticello (Letter from NRC to NMC, "Monticello Nuclear Generating Plant - Second Request For Additional Information Related To Technical Specifications Change Request To Implement A 24-Month Fuel Cycle (TAC No. MC3692)," dated January 31, 2005, (see NRC RAI 1) -- Reference ADAMS ML050180738). The Monticello response was provided in a Letter from NMC to NRC, "Response to NRC Requests for Additional Information Regarding License Amendment Request Supporting 24-Month Fuel Cycles (TAC No. MC3692)," dated March 3, 2005 (Reference ADAMS ML050670432), Enclosure 2, NRC Question #1. The information to this previous response included information that is currently being requested in Question 7 of this Open Item. Therefore, this previous Monticello response is applicable to Question 7 of this Open Item. Furthermore, a sample setpoint calculation was provided in a letter from NMC to NRC, "Supplemental Submittal in Support of 24-Month Fuel Cycles License Amendment Request (TAC No. MC3692)," dated November 5, 2004 (Reference

ADAMS ML043150428). This sample calculation includes how the two separate verifications are performed.

✔ Close

NRC ITS TRACKING

NRC Reviewer

ID	**200512051806** Conference Call Requested? **No**
Category	**Editorial**
ITS Information	**ITS Section:** 3.6 **ITS Number:** 3.6.1.3 **DOC Number:** M.1 **JFD Number:** 9 **Bases JFD Number:** None **Page Number(s):** 80
Comment	**Although the RHR isolation valves are the only PCIVs required operable in Modes 4 & 5, and OPDRVs apparently can only occur in those Modes, to avoid confusion and to maintain standardization, adopt the phrase in ITS Condition F "or during OPDRVs."**
Issue Date	**12/05/2005**

Close Date	**01/17/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/09/2005	Improved Standard Technical Specification (ISTS) 3.6.1.3 Condition H (Attachment 1, Volume 11, Rev. 0, Page 80 of 431) requires entry when "Required Action and associated Completion Time of Condition A, B, C, D, or E not met for PCIV(s) required to be OPERABLE during MODE 4 or 5 or during operations with a potential for draining the reactor vessel (OPDRVs)." ISTS 3.6.1.3 provides bracketed ACTIONS for all shutdown conditions, i.e., "during movement of [recently irradiated fuel assemblies in [secondary] containment" (ISTS 3.6.1.3 Condition G, "during MODE 4 or 5" (ISTS 3.6.1.3 Condition H), and "during operations with a potential for draining the reactor vessel (OPDRVs)" (ISTS 3.6.1.3 Condition H) (Page 80 of 431). These ACTIONS were provided to ensure all potential Applicabilities listed in ISTS 3.3.6.1, "Primary Containment Isolation Instrumentation," were covered. However, the ISTS 3.6.1.3 ACTIONS were bracketed, since a plant-specific Improved Technical Specification (ITS) submittal would not necessarily include all these Applicabilities in ITS 3.3.6.1. In the Monticello ITS, the phrase "or during operations with a potential for draining the reactor vessel (OPDRVs)" in ISTS 3.6.1.3 Condition H has not been included in ITS 3.6.1.3 Condition F . This change has been justified in Justification for Deviation 9 (Page 86 of 431) which states, "The words in ISTS 3.6.1.3 Condition H (ITS 3.6.1.3 Condition F), "or during operations with a potential for draining the reactor vessel (OPDRVs)," have been deleted. There are no PCIVs required to be OPERABLE in the Monticello ITS whose Applicability is only during OPDRVs. The only PCIVs required when not in MODES 1, 2, and 3 are the RHR shutdown cooling isolation valves, and their Applicability is MODES 1, 2, 3, 4 and 5. This Condition is still

	applicable in MODES 4 and 5, which are the only MODES that OPDRVs can be performed. Therefore, the "during OPDRVs" Applicability is duplicative of the MODES 4 and 5 Applicability and has been deleted.? Furthermore, maintaining the term "during OPDRVs" in the Condition, when no ITS 3.3.6.1 instrument Applicability specifies "during OPDRVs" would create more confusion than not including the term. In addition, this change is consistent with the approved ITS Licensing Amendments for FitzPatrick, Quad Cities Units 1 and 2, Dresden Units 2 and 3, LaSalle Units 1 and 2, and Nine Mile Point 2. Therefore, Monticello considers the proposed change to be consistent with the industry and a better presentation of the requirement as it applies to Monticello.

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NRC ITS TRACKING

NRC Reviewer

ID	200512052051 Conference Call Requested? Yes
Category	Discussion
ITS Information	ITS Section: 3.6 ITS Number: 3.6.1.3 DOC Number: A.8 JFD Number: 5 Bases JFD Number: None Page Number(s): 61
Comment	Discuss how the 2 gpm (120 gph) was found acceptable by the NRC in response to the 1973 letter referenced in DOC A8. Provide a copy of the letter and the relevant parts of the NRC staff's approval document. Confirm that the radiological consequences of an EFCV line break event has been analyzed and that the anaysis assumed a 2 gpm leak rate. Include specific reference to FSAR analysis and the letter from 1973 in the Bases for SR 3.6.1.3.8. Briefly, why are the conditions (alluded to by the Reviewer's note) in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000, met? If not, why is 120 gph acceptable?
Issue Date	12/05/2005

Close Date	02/24/2006 Resolution requires change to: NUREG Bases Markup Typed ITS Bases Docket Response Required? Yes

Responses

Licensee Response by Jerry Jones on 12/07/2005	The NRC issued a safety evaluation report (SER) for the Monticello full-term operating license application, dated February 3, 1973. In section 6.2 of the SER, the NRC stated that they calculated the doses resulting from an instrument line break, and concluded that the instrument lines penetrating the reactor containment were acceptable as installed. In addition, the original NRC Technical Specifications for Monticello were issued as part of Monticello's Provisional Operating License (POL). The Surveillance requiring excess flow check valve testing is identical to that in Current Technical Specification (CTS) 4.7.D.1.b. However, the associated Bases for this Surveillance stated "The containment is penetrated by a large number of small diameter instrument lines. The program for the periodic testing (See Specification 4.7.D) and examination of the valves in these lines will be developed and a report covering this program submitted to the Atomic Energy Commission." The report, provided the Monticello excess flow check valve test program (which includes the leakage limit), was submitted in a letter from L. O. Mayer (Northern States Power) to J. F. O'Leary (NRC), dated July 27, 1973. This letter is included as an attachment to this response. The Monticello POL Technical Specifications Bases described above were changed subsequent to the submittal of the excess flow check valve report. The Bases were

	changed to state "The containment is penetrated by a large number of small diameter instrument lines. The program for the periodic testing (See Specification 4.7.D) and examination of the valves in these lines has been developed and a report covering this program was submitted to the AEC on July 27, 1973." This Bases changed was issued to Monticello by the NRC as part of Amendment 22 to the POL, dated July 13, 1976. Therefore, it is the position of Monticello that the NRC has reviewed and accepted the excess flow check valve leakage limit provided in the July 27, 1973 letter. The Improved Technical Specification (ITS) Bases for SR 3.6.1.3.8 (Attachment 1, Volume 11, Rev. 0, Pages 106 and 109 of 431) will be modified to reference the July 27, 1973 letter, as shown in an attachment to this response. Furthermore, the NRC question concerning NEDO-32977-A is not applicable to Monticello. NEDO-32977-A concerns extending the test Frequency for excess flow check valves, which Monticello is not proposing in the ITS submittal.

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Last Modified: 02/24/2006 07:37 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200512091815 Conference Call Requested? No
Category	Major Technical
ITS Information	ITS Section: 3.6 DOC Number: None JFD Number: 1 Page Number(s): 384 ITS Number: None Bases JFD Number: None
Comment	Additional remark: Will need to discuss this comment point by point. Adopt STS 3.6.1.4 and SR 3.3.1.1.1 (CHANNEL CHECK) for drywell pressure and the drywell high pressure trip instrumentation function (3.3.1.1.6), respectively. During plant operation, operators monitor drywell pressure on a shiftly (at least) basis by observing its instrumentation indication in the control room, and in the event of an increasing trend in drywell pressure and before reaching the 2.0 psig trip setpoint, operators would be expected to take action to control pressure to avoid an RPS actuation. Therefore, conforming to the STS in these instances is not an additional burden and is a reasonable expectation. DOC 3.3.1.1-L1 states drywell high prssure trip function is not credited in the accident analysis, however, a drywell initial pressure of 2.0 psig is assumed; hence, there is more justification for adopting STS 3.6.1.4 than just retaining the trip function. Potentially impacts DOCs 3.3.1.1 M6 M8 L1 L5, STS markup page 52 JFD 3.3.1.1 - 12 (all the more reason to adopt STS SR 3.6.1.4.1) STS 3.3.1.1 Bases markup pages 60, 79 Could this affect ITS 3.3.3.1 function 4? Note that ITS SR 3.3.3.1.1 requires a 31-day frequency to perform a Channel Check of Function 4; why can it be done here, but not in 3.3.1.1? Different indications? STS 3.3.3.1 Bases markup page 240 DOCs 3.3.5.1 A3 M6 STS markup pages 349, 351, 353, 355, 356, 359, STS 3.3.5.1 Bases markup pages 362, 363, 364, 365, 366, 367, 369, 372 Note that pages 375, 385, & 387 state that the recirculation line break LOCA credits the drywell pressure - high function. Is this correct? Are the RPS and ECCS drywell pressure instruments different? JFDs 3.3.5.1 - 2, 11 (all the more reason to adopt STS SR 3.6.1.4.1) DOCs 3.3.6.1 M4 M6 (what if drywell pressure instrument sensors isolated?) L8 - Note, this is in fact an LA-type change. JFD 3.3.6.1 - 9 STS markup pages 497, 498, 572 B 3.3.6.1 - markup pages 504 DOCS 3.3.6.2 A3, L3 - Note, this is in fact an LA-type change. JFD 3.3.6.2 - 7 B 3.3.6.2 - markup pages 577

	JFD 3.3.6.2 - 7
Issue Date	12/09/2005

Close Date	03/28/2006	Resolution requires change to: None Docket Response Required? No

▼Responses

Licensee Response by Jerry Jones on 12/12/2005	Improved Standard Technical Specification (ISTS) 3.6.1.4, "Drywell Pressure" (Attachment 1, Volume 11, Rev. 0, Page 383 of 431), was not adopted in the Monticello Improved Technical Specification (ITS) as discussed in Justification for Deviation (JFD) 1 (Page 384 of 431). JFD 1 states "ISTS 3.6.1.4 has not been adopted since it is not applicable to Monticello. The Monticello containment analyses assume an initial drywell pressure of 2.0 psig. However, if the pressure in the drywell were to reach this pressure, the unit will automatically shutdown because the Reactor Protection System Drywell Pressure - High Allowable Value is less than or equal to 2.0 psig. Therefore, ISTS 3.6.1.4 is not needed to ensure drywell pressure is within limit because ITS 3.3.1.1, Reactor Protection System (RPS) Instrumentation, will ensure the drywell pressure remains less than or equal to 2.0 psig. This is also consistent with current licensing basis since Monticello does not include a specific requirement to maintain drywell pressure within a specified limit." This type of justification has been previously used to delete a CTS Drywell Pressure requirement during the Washington Public Power Supply System Unit 2 (i.e., Columbia Generating Station) ITS conversion (NRC Safety Evaluation Report for ITS Amendment 149, dated 3/4/97, pages 195 and 196). This type of justification has also used for other variables in the ISTS where an RPS Function provided automatic protection. Specifically, in the FitzPatrick ITS conversion, ISTS 3.4.10, "Reactor Steam Dome Pressure," was not adopted for similar reasons as described in ISTS 3.6.1.4 JFD 1. In addition, while the drywell pressure limit is not being proposed in the Monticello ITS, operators currently monitor various parameters during plant operation and, if the parameters are not within pre-determined limits, take appropriate action based on plant operating procedures. This includes the drywell pressure parameter. However, there is no need to add a new Technical Specification to ensure the appropriate monitoring occurs. Furthermore, while ITS Table 3.3.1.1-1, Function 6, Drywell Pressure - High is not credited in the accident analysis, it is being retained in the Monticello ITS and will ensure a reactor scram occurs if drywell pressure reaches or exceeds 2.0 psig, thus preserving the accident analysis assumptions. In addition, the CHANNEL CHECK (SR 3.3.1.1.1, Page 45 of 760) associated with ITS Table 3.3.1.1-1, Function 6 Drywell Pressure - High (Page 52 of 760) was not adopted as discussed in ITS 3.3.1.1 JFD 12 (Page 56 of 760) (the drywell pressure instruments for the RPS do not have indicators, only switches). Based on the above, Monticello will not adopt ISTS 3.6.1.4. Furthermore, the NRC reviewer's comment has identified other potential impacts to Instrumentation Specifications related to the drywell pressure limit. Monticello does not see any conflicts or any impact to the specified Instrumentation Specifications.

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200512091847** Conference Call Requested? **Yes**
Category	**Major Technical**
ITS Information	**ITS Section:** **3.6** **ITS Number:** **None** **DOC Number:** **None** **JFD Number:** **1** **Bases JFD Number:** **None** **Page Number(s):** **402**
Comment	**Adopt STS 3.6.2.5; will refer to ACVB for review. The STS Bases gives another reason for the LCO besides limiting torus loading - Its effect on the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident LOCA. Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).**
Issue Date	**12/09/2005**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 12/12/2005	Improved Standard Technical Specification (ISTS) 3.6.2.5, "Drywell-to-Suppression Chamber Differential Pressure" (Attachment 1, Volume 11, Rev. 0, Page 401 of 431), was not adopted in the Monticello Improved Technical Specification (ITS) as discussed in Justification for Deviation (JFD) 1 (Page 402 of 431). JFD 1 states "ISTS 3.6.2.5 has not been adopted since it is not applicable to Monticello. The Monticello containment analyses for a DBA LOCA do not assume a drywell-to-suppression chamber differential pressure to reduce the hydrodynamic loads on the torus during a LOCA blowdown. Therefore, ISTS 3.6.2.5 is not needed to ensure a drywell-to-suppression chamber differential pressure limit. This is consistent with the current Monticello licensing basis." The wording in JFD 1 concerning the hydrodynamic loads was used since this is the reason given in the ISTS 3.6.2.5 Bases for the reason for the Technical Specification requirement. Specifically, ISTS 3.6.2.5 Applicable Safety Analysis Bases (Page 404 of 431), first paragraph states that "The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown." Thus, minimizing the peak drywell pressure during downcomer pipe clearing during a DBA LOCA is one of the factors in reducing the hydrodynamic loads (the other is minimizing the mass

	of the accelerated water leg). The Monticello Updated Safety Analyses Report (USAR) Table 5.2-7, "Assumptions for the LOCA Containment Evaluation," states that both the drywell and wetwell pressure were initially at the same pressure, 16.7 psia, for the containment analyses (except for net positive suction head cases, where 14.26 psia is assumed). Therefore, since no differential pressure between the drywell and suppression chamber was assumed in the analysis, there is not need to include a specific Specification on drywell-to-suppression chamber differential pressure. This is consistent with the current Monticello licensing basis.

Date Created: 12/09/2005 06:47 PM by Craig Harbuck
Last Modified: 03/28/2006 05:11 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200512121613 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.8 ITS Number: 3.8.7 DOC Number: None JFD Number: 3 Bases JFD Number: None Page Number(s):
Comment	**Please provide justification for deleting iSTS LCO 3.8.9, Required Action B, "One or more AC vital buses inoperable."**
Issue Date	12/12/2005

Close Date	01/30/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 12/14/2005	Improved Standard Technical Specification (ISTS) LCO 3.8.9 (Improved Technical Specifications (ITS) LCO 3.8.7) (Attachment 1, Volume 13, Rev. 0, Page 234 of 294) states "[Division 1] and [Division 2] AC, DC, [and AC vital bus] electrical power distribution subsystems shall be OPERABLE." In addition, ISTS 3.8.9 ACTION B, which is also bracketed, provides requirements for inoperable AC vital buses. The requirements for AC vital buses in ISTS 3.8.9 have not been included in Monticello ITS 3.8.7. The AC vital bus requirements were deleted as justified in Justification for Deviation (JFD) 3 (Page 236 of 294). JFD 3 states "The bracketed AC vital bus requirements have not been adopted in the Monticello ITS. The change is consistent with the current requirements in the current licensing basis. Subsequent ACTIONS have been renumbered, as necessary." ISTS 3.8.9 (ITS 3.8.7) provides the requirements for the electrical power distribution subsystems. The ISTS 3.8.9 requirements for the AC vital buses are all in brackets. This indicates that the AC vital bus requirement is plant-specific. Monticello does not have any AC vital bus requirements in the Current Technical Specifications (CTS). If an AC vital bus is inoperable (i.e., de-energized), the CTS definition of OPERABILITY requires the affected components to be declared inoperable (since it has no power), and the appropriate CTS Actions for the inoperable component taken. In the conversion to the ITS, Monticello has chosen to maintain the currently required list of AC buses (i.e., the buses that are required by CTS 3.9.A.3 (Page 227 of 294). As in the CTS, if an AC vital bus becomes de-energized, the affected components will be declared inoperable (as required by the ITS definition of OPERABILITY) and appropriate ITS ACTIONS will be taken. Therefore, the justification provided in JFD 3 ("...change is consistent with current requirements in the current licensing basis.")

	is an accurate and correct justification for not adopting the ISTS AC vital bus requirements. Furthermore, ITS 3.8.7 Bases, LCO section, INSERT 2 (Page 242 of 294) reiterates the requirement that if a bus not listed in Table B 3.8.7-1 (the Table listing the AC and DC buses required by ITS LCO 3.8.7) becomes inoperable (i.e., de-energized), and its de-energization is not the result of an inoperability of a bus listed in Table 3.8.7-1, then the individual loads on the de-energized bus will be declared inoperable and the appropriate Conditions and Required Actions of the LCOs governing the individual loads will be entered.

Date Created: 12/12/2005 04:13 PM by Robert Clark
Last Modified: 01/30/2006 05:20 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200512151123 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.4 ITS Number: 3.4.3 DOC Number: LA.3 JFD Number: None Bases JFD Number: None Page Number(s): 54
Comment	**In a phone call on 12/15/2005, Monticello indicated LA3 was to be changed. Please provide details of change.**
Issue Date	12/15/2005

Close Date	02/28/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 12/16/2005	Discussion of Change (DOC) LA.3 (Attachment 1, Volume 9, Rev. 0, Page 54 of 255) proposes to relocate Current Technical Specification (CTS) 4.6.E.1.b (Page 5 of 255) to the Inservice Testing (IST) Program. CTS 4.6.E.1.b requires that at least two of the safety/relief valves be disassembled and inspected each refueling interval. This requirement was proposed to be relocated to the IST Program since this is the program that controls S/RV testing per the ASME Operation and Maintenance (OM) Code. However, the Monticello IST Program does not currently include this CTS requirement; it is located in a maintenance procedure. Therefore, instead of relocating the CTS requirement to the IST program, the requirement will be relocated to the Technical Requirements Manual (TRM). This is acceptable because changes to the TRM will be made under 10 CFR 50.59, which ensures changes to the TRM are properly evaluated. DOC LA.3 will be changed to reflect this new location (i.e., the TRM), as shown in the attachment to this response.
NRC Response by David Roth on 02/28/2006	No questions on revised LA3.

Date Created: 12/15/2005 11:23 AM by David Roth
Last Modified: 02/28/2006 10:41 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200512151125** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **None** ITS Number: **None** DOC Number: **None** JFD Number: **None** Bases JFD Number: **None** Page Number(s): **1**
Comment	**Please provide details of any recently-approved TSTFs to be incorporated into ITS.**
Issue Date	**12/15/2005**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/20/2006	TSTF-372, Rev. 4 has been approved since the cut-off date specified in the Monticello ITS submittal transmittal letter, dated 6/29/05. This TSTF will be incorporated into the Monticello ITS as shown in the attachment to this response.
Licensee Response by Jerry Jones on 01/05/2006	The NRC Technical Specifications Branch Chief (T. H. Boyce) informed the Technical Specifications Task Force (TSTF), by letter dated 12/6/05, that TSTF-343, TSTF-479, TSTF-482, and TSTF-485 would be incorporated into revision 3.1 of the Improved Standard Technical Specifications (ISTS). Monticello has reviewed and evaluated these approved TSTFs. TSTF-343, Rev. 1: This TSTF will not be adopted in the Monticello Improved Technical Specifications (ITS) as it is not applicable to the Monticello design (Monticello does not have pre-stressed concrete containment tendons in the primary containment, as stated in ITS 5.5, Justification For Deviations (JFD) 7 (Attachment 1, Volume 17, Rev. 0, Page 97 of 143). TSTF-479, Rev. 0: This portion of the TSTF concerning the change from Boiler and Pressure Vessel Code, Section XI to Operation and Maintenance (OM) Code has already been incorporated into the Monticello ITS. While the wording in the Monticello ITS is not identical to the TSTF, Monticello has changed the applicable references from ASME Boiler and Pressure Vessel Code, Section XI to the ASME OM Code. Therefore, changes provided by this TSTF related to the OM Code are not required in the Monticello ITS. However, the portion of the change related to Improved Standard Technical Specifications (ISTS) 5.5.7.b (ITS 5.5.5.b), i.e., the modification to when SR 3.0.2 applies, is not currently in the Monticello ITS. Therefore, the Monticello ITS submittal will be revised to include this part of the TSTF, as shown in the attachment to this response. TSTF-482, Rev. 0: This TSTF will be adopted. It should be noted that the second and third changes provided by the TSTF are already incorporated into the Monticello ITS with a JFD (JFD 3) that states that a

	typographical/grammatical error has been corrected (Attachment 1, Volume 5, Rev. 0, Page 49 of 63). The Monticello ITS submittal will be revised to include the first change of the TSTF, and a new justification for all three changes. The changes are provided in the attachment to this response. TSTF-485, Rev. 0: This TSTF will be incorporated into the Monticello ITS. The changes are provided in the attachment to this response.
NRC Response by David Roth on 02/09/2006	Please be aware the TSTF-479 reviews are currently (as of 02/06/2006) suspended to clarify the Traveler's applicability to greater-than-two-year (that is 5-year and 10-year) frequencies. Suggest not incorporating TSTF-479 at this time to avoid any associated delays.
Licensee Response by Jerry Jones on 02/13/2006	As stated in the first Monticello response to this RAI (Monticello response on 1/5/06), Monticello had already adopted the change from Boiler and Pressure Vessel Code, Section XI to Operation and Maintenance (OM) Code, based on our current licensing basis. Furthermore, our response stated that the portion of the Technical Specifications Task Force (TSTF) -479 change related to Improved Standard Technical Specifications (ISTS) 5.5.7.b (ITS 5.5.5.b), i.e., the modification to when SR 3.0.2 applies, is not currently in the Monticello ITS and that it would be added into the ITS. The appropriate changes were provided in the attachment to the first Monticello response. However, after Monticello provided the first response and appropriate changes to adopt TSTF-479, the NRC recommended that Monticello not incorporate this part of the TSTF since it might delay the approval of the ITS. Based on this NRC recommendation, Monticello will not adopt the TSTF-479 change associated with ISTS 5.5.7.b (ITS 5.5.5.b). Therefore, last five pages of the attachment to the Monticello response of 1/5/06, which are all related to TSTF-479 (and are stamped at the top left as being "TSTF-479 related"), will not be included in the future revision to the Monticello ITS submittal. These five pages show the change to ITS 5.5.5.b and associated Current Technical Specifications Markup, Discussion of Changes, and Justification for Deviations changed pages.
NRC Response by David Roth on 02/28/2006	For TSTF-372, Rev. 4, note that it has been CLIIPED and a model application posted on the NRC's website at http://www.nrc.gov/reactors/operating/licensing/techspecs/changes-issued-for-adoption.html Please fill in the blanks in the TSTF-372 Rev. 4 CLIIP model and submit it as a response to this question. At a minumum, fill in this section: 2.0 ASSESSMENT 2.1 Applicability of Published Safety Evaluation [LICENSEE] has reviewed the safety evaluation dated [DATE] as part of the CLIIP. This review included a review of the NRC staff?s evaluation, as well as the supporting information provided to support TSTF-372. [LICENSEE] has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to [PLANT, UNIT NOS.] and justify this amendment for the incorporation of the changes to the [PLANT] TS. 2.2 Optional Changes and Variations [LICENSEE] is not proposing any variations or deviations from the TS changes described in the TSTF-372 Revision 4 or the NRC staff?s model safety evaluation dated [DATE].

Date Created: 12/15/2005 11:25 AM by David Roth
Last Modified: 03/28/2006 07:19 AM

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NRC ITS TRACKING

NRC Reviewer

ID	200512161429 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: None DOC Number: None JFD Number: None Bases JFD Number: None Page Number(s):
Comment	This question regards recent correspondence between the NRC, the NEI Setpoint Methods Task Force (SMTF). On June 29, 2005, Monticello submitted its License Amendment Request for Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS) On July 1, 2005, NMC committed to continuing to reset Limiting Safety System Settings (LSSS) setpoints within the specified tolerances (as-left criteria) until the Technical Specification Task Force's TS change pertinent to instrument setpoints was approved by the NRC and assessed for applicability to MNGP. NMC also committed to assessing applicability of the Technical Specification Task Force's TS change pertinent to instrument setpoints, when approved by the NRC, to determine whether changes to MNGP's licensing basis are necessary. (ref: ADAMS ML051890051) By letter dated September 7, 2005, from Patrick Hiland, Chief, Reactor Operations Branch, to the NET SMTF, (ADAMS ML052500004), the NRC provided draft changes to plant TS that are acceptable to the NRC staff for implementing the concepts related to setpoint allowable values for safety-related instrumentation. The letter provided a checklist for development of TS using SMTF agreement concepts, and included notes for incorporation into LCO surveillances or footnotes to surveillances. The checklist was intended for review of plant-specific license amendment requests for changes to TS setpoint Allowable Values. Please indicate what, if any, changes to Monticello's proposed ITS will be made based on the information in the September 7, 2005 letter.
Issue Date	12/16/2005

Close Date	03/09/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 01/11/2006	The NRC letter from Patrick Hiland (NRC) to NEI Setpoint Methods Task Force dated 9/7/05 has been an input into the Technical Specifications Task Force (TSTF) traveler being developed. This TSTF is currently scheduled to be provided to the NRC in mid-January, 2006. As stated in the NRC question, NMC has already committed to assessing applicability of the TSTF traveler when approved by the NRC. Therefore, no specific changes will be made to the

	Monticello ITS based solely on the NRC letter of 9/7/05.

Date Created: 12/16/2005 02:29 PM by David Roth
Last Modified: 03/09/2006 01:06 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200601061257 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: 3.3.1.2 DOC Number: None JFD Number: 5 Bases JFD Number: None Page Number(s): 131
Comment	Per phone call on 01/06/2006 from Roger Scott, Monticello, to David Roth, NRC, a correction to proposed Note 1 is required. Please provide the correction.
Issue Date	01/06/2006

Close Date	03/28/2006 Resolution requires change to: JFD Typed ITS Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 01/11/2006	ITS SR 3.3.1.2.5 (Attachment 1, Volume 8, Rev. 0, Page 127 of 760) requires the performance of a CHANNEL FUNCTIONAL TEST and determination of a signal to noise ratio. This Surveillance is modified by a Note (Page 128 of 760) that states "Not required to be met with less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant." However, Justification for Deviation (JFD) 5 (Page 131 of 760) and the Bases discussion for the Note to SR 3.3.1.2.5 (Page 140 of 760) both state that this Note only applies to the determination of signal to noise ratio portion of the SR. Therefore, this Note will be modified to state ?The determination of signal to noise ratio is not required to be met with less than or equal to two fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant.? This change is shown in the attachment to this response. This change is also consistent with TSTF-455, a "T" type traveler.

Date Created: 01/06/2006 12:57 PM by David Roth
Last Modified: 03/28/2006 07:16 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200601102101** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	ITS Section: **3.6** ITS Number: **3.6.1.7** DOC Number: **LA.3** JFD Number: **None** Bases JFD Number: **None** Page Number(s):
Comment	**Regarding the removal of the vacuum breaker visual inspection cyclic surveillance requirement in CTS 4.7.A.4.a(2). Explain how 50.55a includes 'visual inspection' of the 8 torus-drywell vacuum breakers, and how this requirement will be specified in the Monticello ITS program. (Aren't these valves already included in the IST Program?)**
Issue Date	**01/10/2006**

Close Date	**03/28/2006** Resolution requires change to: DOC Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/24/2006	Current Technical Specifications (CTS) 4.7.A.4.a.(2) (Attachment 1, Volume 11, Rev. 0, Page 172 of 431), in part, requires the pressure suppression chamber-drywell vacuum breakers to be visually inspected once each operating cycle. This requirement has been proposed to be relocated to the Inservice Testing (IST) Program as described in Discussion of Change (DOC) LA.3 (Page 178 of 431). The pressure suppression chamber-drywell vacuum breakers are included in the Monticello IST Program. The Monticello IST Program specifies test requirements for these vacuum breakers: Category A Seat Leakage Test, Category C Exercise Test Open, Category C Exercise Test Closed, and Position Indication Test. During the performance of the Category C exercise tests (Open and Close) and the position indication test, an individual is required to be at the vacuum breakers, visually ensuring each vacuum breaker opens and closes and confirming proper operation of the position indication. Since the vacuum breakers are visually inspected for proper operation during these three tests and CTS 4.7.A.4.a.(2) does not describe the acceptance criteria for the required visual inspection, it is Monticello's position that the visual inspections required for these three tests can be used to meet the CTS 4.7.A.4.a.(2) requirement. However, based on the NRC reviewers comment, NMC has re-evaluated the proposed relocation to the IST Program and has decided that, for clarity, a specific visual inspection requirement for the vacuum breakers should be maintained in lieu of relying solely upon the above described visual inspection that is part of the three IST test requirements (Category C exercise test open and close and position indication test). Therefore,

	DOC LA.3 will be modified to propose the visual inspection requirement of CTS 4.7.A.4.a.(2) be relocated to the Technical Requirements Manual (TRM). This is acceptable because changes to the TRM will be made under 10 CFR 50.59, which ensures changes to the TRM are properly evaluated. Specific changes to DOC LA.3 are shown in the attachment to this response.

Date Created: 01/10/2006 09:01 PM by Craig Harbuck
Last Modified: 03/28/2006 10:23 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200601111646** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	**ITS Section:** 3.6 **ITS Number:** 3.6.1.8 **DOC Number:** A.2 **JFD Number:** 1 **Bases JFD Number:** None **Page Number(s):** 206
Comment	**Verify that IST SR 3.6.2.3.2 for the RHR pumps in the RHR suppression pool cooling Specification is adequate for demonstrating RHR pump operability to support the operability of the associated drywell spray subsystem, and that failure of that SR may require entering appropriate Actions in ITS 3.6.1.8 as well as ITS 3.6.2.3 - this would require amending the ITS 3.6.1.8 Bases for Action A to point this out, or better yet, a note in Action A of ITS 3.6.2.3; else, include STS SR 3.6.2.4.2 in ITS 3.6.1.8 (suitably modified to indicate drywell spray instead of suppression pool spray).**
Issue Date	**01/11/2006**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **Yes**

Responses

Licensee Response by Jerry Jones on 01/13/2006	SR 3.6.2.3.2 (Attachment 1, Volume 11, Rev. 0, Page 268 of 431) requires the verification that each required Residual Heat Removal (RHR) pump develops a flow rate greater than 3870 gpm through the associated heat exchanger while operating in the suppression pool cooling mode. This is the Surveillance Requirement to ensure the RHR pump provides adequate flow while in the suppression pool cooling mode. The Monticello accident analysis, with respect to the required flow rate while in the drywell spray mode, assumes flow is 3800 gpm. Furthermore, the Current Technical Specifications (CTS) does not require a flow rate test for the drywell spray mode, only a test to ensure the spray nozzle and header is unobstructed (CTS 4.5.C.1) (Page 200 of 431). This test is maintained in the Improved Technical Specifications (ITS) as SR 3.6.1.8.2 (Page 206 of 431). The format of the BWR Improved Standard Technical Specifications (ISTS) does not provide cross-references to other Technical Specifications for equipment that has requirements in multiple Technical Specifications. For example, ITS 3.4.7 provides the requirements for the RHR Shutdown Cooling System when in Mode 3 with reactor steam dome pressure less than the RHR shutdown cooling supply isolation interlock. The RHR Shutdown Cooling System uses the same RHR pumps to meet the requirements as are required by ITS 3.6.1.8. However, ITS 3.4.7 does not have any flow rate Surveillances for all the required RHR pumps, and does not provide any cross-references to any other

	RHR pump-related Specifications or Surveillances. If SR 3.6.2.3.2 were not met due to an inoperable RHR pump, ITS 3.4.7 could be affected if ITS 3.4.7 was crediting the RHR pump that has been found inoperable during the performance of SR 3.6.2.3.2. The ISTS does not provide cross-references, but relies on the definition of OPERABILITY to ensure that ITS LCO 3.4.7 would be declared not met in this case. Furthermore, failure of SR 3.6.2.3.2 may be met for a reason unrelated to the RHR pumps capability to provide proper drywell spray flow. For example, it may be due to failure of an RHR suppression pool valve, which would not impact the capability of the RHR pump to provide adequate drywell spray flow. The Bases of ITS 3.6.1.8 (Page 211 of 431) states that each RHR drywell spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping (including drywell spray header and nozzles), valves, instrumentation, and controls are OPERABLE. This Bases statement and the training and knowledge of plant personnel will ensure the appropriate Conditions and Required Actions of ITS 3.6.1.8 are entered when an RHR pump cannot achieve the required drywell spray flow requirements, even if this is discovered during performance of an RHR pump test in another Specification (i.e., ITS 3.5.1 (the Emergency Core Cooling System Specification, which also includes RHR pump flow tests) or ITS 3.6.2.3). Therefore, the Monticello ITS submittal does not need to be modified to include any cross-references to other Specifications that also have RHR pump requirements. This is consistent with previous ITS conversions for other BWR designs that have similar requirements. The approved ITS conversions for both Quad Cities 1 and 2 and Dresden Units 1 and 2 include an RHR Suppression Pool Cooling Specification (ITS 3.6.2.3) and an RHR Suppression Pool Spray System (ITS 3.6.2.4) Specification. For both these plants, ITS 3.6.2.3 included a flow test while ITS 3.6.2.4 did not include a flow rate test (a test to ensure the suppression pool spray nozzles were not unobstructed was included, similar to the Monticello ITS submittal). In addition, neither the Specifications nor the Bases were modified to explicitly cross-reference the two Specifications.

Date Created: 01/11/2006 04:46 PM by Craig Harbuck
Last Modified: 03/28/2006 10:29 AM

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NRC ITS TRACKING

NRC Reviewer

ID	**200601111847** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	**ITS Section:** 3.6 **ITS Number:** 3.6.2.3 **DOC Number:** None **JFD Number:** 2 **Bases JFD Number:** None **Page Number(s):** 268
Comment	**Only one RHR pump is required for an operable subsystem of RHR suppression pool cooling; hence ITS SR 3.6.2.3.2 requires testing each 'required' RHR pump. The same RHR pumps are also tested as part of the LPCI subsystem by ITS SR 3.5.1.7. Describe how these SRs overlap, and thus why using the word 'required' in SR 3.6.2.3.2 is needed?**
Issue Date	**01/11/2006**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/16/2006	Improved Technical Specification (ITS) SR 3.6.2.3.2 (Attachment 1, Volume 11, Rev. 0, Page 268 of 431) requires the verification that each "required" RHR pump develops a flow rate greater than 3870 gpm through the associated heat exchanger while operating in the suppression pool cooling mode. The LCO section of the Bases for ITS 3.6.2.3 (Page 272 of 431) states that only "one" of the RHR pumps in each subsystem is required to be OPERABLE. In the Improved Standard Technical Specifications (ISTS), when there are more installed components in the plant than are required by the LCO, the word "required" is used in ACTIONS and Surveillance Requirements for clarity. For example, ISTS 3.6.3.1 uses the word "required" in SR 3.6.3.1.1 and 3.6.3.1.2 (Page 411 of 431) since the Bases states that there are more drywell cooling system fans installed than are required by the LCO (Pages 414 and 415 of 431). Therefore, the word "required" has been included in ITS SR 3.6.2.3.2, as justified in Justification for Deviation (JFD) 2 (Page 269 of 431). ITS LCO 3.5.1 (Attachment 1, Volume 10, Rev. 0, Page 23 of 118) requires each Emergency Core Cooling System (ECCS) injection/spray subsystem to be OPERABLE. The Background section of the Bases for ITS 3.5.1 (Page 37 of 118) states that there are two LPCI subsystems, each consisting of two motor driven pumps in the same RHR loop and piping and valves to transfer water from the suppression pool to the RPV via the selected recirculation loop. The LCO section of the Bases (Page 41 of 118) states that the ECCS injection/spray subsystems are defined as the "two LPCI subsystems." Based on the discussion in the Bases Background and LCO sections, all RHR pumps are required to be OPERABLE by ITS 3.5.1. Thus, ITS SR 3.5.1.7 (Page 30 of 118)

	requires all of the RHR pumps to develop the specified flow rate. The word "required" is not included in this Surveillance Requirement, since all the RHR pumps are required by ITS 3.5.1. If "one" RHR pump does not satisfy the flow requirements in ITS SR 3.5.1.7 and ITS SR 3.6.2.3.2, ITS 3.5.1 is not met and the appropriate ACTION must be entered. However, the ACTIONS of ITS 3.6.2.3 (Page 267 of 431) do not have to be entered since all the pumps are not required to be OPERABLE and the LCO requirements are still being met. If another RHR pump in the same loop were to become inoperable, then the ACTIONS of ITS 3.6.2.3 would be required to be entered. Therefore, the word "required" is used correctly in ITS SR 3.6.2.3.2 and must be included in the Surveillance.

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NRC ITS TRACKING

NRC Reviewer

ID	**200601121745** Conference Call Requested? **No**
Category	**Major Technical**
ITS Information	ITS Section: **3.6** ITS Number: **3.6.4.1** DOC Number: **L.1** JFD Number: **None** Bases JFD Number: **None** Page Number(s): **303**
Comment	**Request telcon to discuss the specific heavy load analysis for the fuel cask movement in the reactor building, such as weight, crane used, path, and the explicit language in the USAR Section 12.2.5 that covers it. Isn't this really an LA-type change because the controls for this activity are controlled by 50.59? Note, this comment also applies to ITS 3.6.4.2 L.1 and 3.6.4.3 L.3**
Issue Date	**01/12/2006**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/23/2006	This issue was discussed at the weekly NRC/Monticello phone conference where updates on the ITS conversion are provided. The specific load analysis discussion in USAR Section 12.2.5, Rev. 22 was discussed, as the NRC stated that they had only Rev. 21 of the USAR. Specifically, Monticello stated that the weight of heavy loads was 1100 lbs. over the spent fuel pool and the reactor core and 1500 lbs. in other areas of the plant, as stated in USAR Section 12.2.5.1. The cranes used are described in USAR Section 12.2.5.2 and the safe load path requirements are listed in USAR Section 12.2.5.3. USAR Section 12.2.5, Rev. 22 is provided as an attachment to this response. Monticello has proposed to delete the Current Technical Specification (CTS) requirements for the Secondary Containment, Secondary Containment Isolation Valves, and Standby Gas Treatment System to be OPERABLE when a fuel cask is being moved within the reactor building and the associated ACTIONS (CTS 3.7.C.2.c, CTS 3.7.C.4.b.3, CTS 3.7.B.1, CTS 3.7.B.1.c.1)(b), CTS 3.7.B.1.c.2)(a), and CTS 3.7.B.1.d) (Attachment 1, Volume 11, Rev. 0, Pages 295, 296, 322, 323, 353, and 355 of 431). The deletion of these requirements is provided in an "L" type Discussion of Change (DOC), in lieu of an "LA" type DOC, since the specific requirements of OPERABILITY and the associated ACTIONS if the above systems are inoperable are not proposed to be maintained in the USAR. It is Monticello's intention to control these requirements through the use of plant procedures that control movement of heavy loads. However, since the NRC has not recently allowed an "LA" type DOC to be used to relocate requirements to plant procedures, Monticello has classified this change

	as an "L" type change. This is consistent with the most recently approved ITS conversion, D.C. Cook, which deleted cask handling ACTIONS from the Spent Fuel Storage Pool Specification (NRC Safety Evaluation for D.C. Cook Units 1 and 2 License Amendments 287 and 269, dated 6/1/05). Therefore, Monticello believes that no changes to the ITS submittal are necessary, and use of an "L" type DOC for this change is acceptable.

Date Created: 01/12/2006 05:45 PM by Craig Harbuck
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NRC ITS TRACKING

NRC Reviewer

ID	**200601171106**		Conference Call Requested? **No**	
Category	**Editorial**			
ITS Information	ITS Section: **3.6** ITS Number: **3.6.4.3**	DOC Number: **None**	JFD Number: **None** Bases JFD Number: **1**	Page Number(s): **376**
Comment	**Bases reference typographical error; USAR 5.4.3.1 should be USAR 5.3.4.1**			
Issue Date	**01/17/2006**			

Close Date	**03/28/2006**	Resolution requires change to: NUREG Bases Markup Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 01/23/2006	Improved Technical Specifications (ITS) 3.6.4.3, Bases Reference 2 (Attachment 1, Volume 11, Rev. 0, Page 376 of 431), will be changed from "USAR, Section 5.4.3.1" to "USAR, Section 5.3.4.1." This change is shown in the attachment to this response.

Date Created: 01/17/2006 11:06 AM by Craig Harbuck
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NRC ITS TRACKING

NRC Reviewer

ID	200601311124 Conference Call Requ
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: 3.3.1.1 DOC Number: None JFD Number: None Bases JFD Number: None
Comment	The RAI requests clarification of information in a phone call and previously posted regardi Per phone call 1/27/2006 between David Roth, NRC, and Roger Scott, MNGP ITS Project M only considered the IRM Neurton Flux high-high change from 120% to 122%, and the ARI +65.6/120% to +67.6/122% to be changes to LSSS. The addition of an allowable value pf 167.8 psig for TCV Fast Closure was not considered l In its answer posted at URL http://www.excelservices.com/exceldbs/itstrack_monticello.nsf/1fddcea10d3bdbb585256c85 SFILE/Monticello%20Response%20to%20RAI%20200510281243%20question%202.pdf, IRM - High High Function does not meet the criteria in 10 CFR 50.36(c)(2)(ii) for being an Please clarify if IRM High High change from 120% to 122% is or isn't LSSS. Please also va posted answer.
Issue Date	01/31/2006

Close Date	03/28/2006	Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 02/10/2006	The second Monticello response provided to RAI 200510281243, which discusses whether or not the IRM - High High Function is an LSSS, provides the correct response; the IRM - High High Function is not considered an LSSS for Monticello. Furthermore, the response incorrectly used a reference to 10 CFR 50.36(c)(2)(iii). It should have been just 10 CFR 50.36(c)(1)(ii)(A). Furthermore, the Monticello response to whether or not the TCV Fast Closure Function is an LSSS will be provided in a subsequent response.
Licensee Response by Jerry Jones on 02/22/2006	As stated in the Applicable Safety Analyses, LCO, and Applicability Sections of the Improved Technical Specifications (ITS) Bases (Attachment 1, Volume 8, Rev. 0, Page 83 of 760), fast closure of the turbine control valves (TCVs) results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Function is the primary scram signal for the generator load rejection without bypass event. For this event, the reactor scram reduces the

	amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded. Thus, the Function does protect against violating a Safety Limit. However, the actual trip setpoint for this Function is not an input into the safety analysis for the generator load reject without bypass event. The analysis assumes the scram signal is input into the RPS within a certain response time. This maximum assumed time response is the time when the acceleration relay assumes control to the time the pressure switches supply the scram trip signal to the reactor protection system logic channels. On the rejection of full load, the oil pressure is dumped from the acceleration relay to cause the control valves to close sooner than they would as operated by the normal governing system. The acceleration relay oil pressure must decay to effectively 0 psig before the acceleration relay moves to cause a fast closure. Since the acceleration oil pressure switch sends a signal at a much higher value (the Allowable Value for the Function is 167.8 psig), the actual setpoint is conservative with respect to when the control valves start to close. In addition, the Monticello Current Technical Specifications (CTS) did not provide a trip setpoint for this Function. CTS Table 3.1.1, Trip Function 11 (Page 9 of 760) does not provide a value for the limiting trip setting for the TCV Fast Closure. It references Note 7 (Page 10 of 760), which states "Trips upon loss of oil pressure to the acceleration relay." Therefore, NMC does not consider the Allowable Value for the TCV Fast Closure, Acceleration Relay Oil Pressure - Low Function to be an LSSS.
Licensee Response by Jerry Jones on 03/12/2006	CTS Table 3.1.1 Function 11 Note 7 states that Turbine Control Valve Fast Closure Function trips upon loss of oil pressure to the acceleration relay. This Note is being replaced with an Allowable Value to be consistent with the format of the Reactor Protection System instrumentation table within NUREG-1433. This Allowable Value is therefore consistent with the current licensing basis and the setpoint methodology used to establish trip settings.

Date Created: 01/31/2006 11:24 AM by David Roth
Last Modified: 03/28/2006 07:17 AM

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NRC ITS TRACKING

NRC Reviewer

ID	200603161318 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: 3.3 DOC Number: None JFD Number: None Bases JFD Number: None Page Number(s):
Comment	Please identify which items were originally considered as BSIs, but actually result in no operational change to sepoints or allowable values when compared to the current license. In other words, identify the items which involved no new calculations, but instead would merely result in listing a different parameter in the ITS, but the value for the parameter came from the same calculations used to provide the information currently in the CTS.
Issue Date	03/16/2006

Close Date	03/28/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 03/16/2006	The Allowable Values for the following BSI items are consistent with the current licensing basis: 1. BSI 1.a: CTS Table 3.1.1 provides trip settings for Function 3.a (Neutron Flux Intermediate Range Monitor (IRM) - High High), Function 4.a (Flow Referenced Neutron Flux Average Power Range Monitor (APRM) High High), and Function 4.c (Flow Referenced Neutron Flux APRM - High Flow Clamp). These trip settings are being replaced with Allowable Values to be consistent with the format of the Reactor Protection System instrumentation table within NUREG-1433. These Allowable Values are therefore consistent with the current licensing basis and the setpoint methodology used to establish the trip settings. 2. BSI 9: CTS Table 3.2.3 provides trip settings for Function 4.b (RBM Downscale). This trip setting is being replaced with an Allowable Value to be consistent with the format of the Control Rod Block instrumentation table within NUREG-1433. This Allowable Value is therefore consistent with the current licensing basis and the setpoint methodology used to establish the trip settings. 3. BSI 1.b: CTS Table 3.2.5 provides trip settings for Function 1 (ATWS-RPT High Reactor Dome Pressure). This trip setting is being replaced with an Allowable Value to be consistent with the format of the ATWS-RPT instrumentation within NUREG-1433. This Allowable Value is therefore consistent with the current licensing basis and the setpoint methodology used to establish the trip settings. 4. BSI 1.g: CTS Table 3.2.1 provides trip settings for Functions 1.b ((High Flow in Main Steam Line), 1.d (Low Pressure in Main Steam Line), and 5.c (Low Pressure in RCIC Steam Supply Line). These trip settings are being replaced with

	Allowable Values to be consistent with the format of the Primary Containment Isolation instrumentation table within NUREG-1433. These Allowable Values are therefore consistent with the current licensing basis and the setpoint methodology used to establish the trip settings. Note that this BSI also covers two other CTS Table Instruments (Functions 5.a and 6.a) whose Allowable Values are being changed (i.e., they are not consistent with current licensing basis). Based on the above and a phone conversation with the NRC reviewers, the Administrative Discussion of Changes (DOC) that describes the terminology change from Trip Settings to Allowable Values in the ITS 3.3 Specifications that also change a value in the CTS Tables will be modified to clearly state that the change is only a terminology change. The proposed Administrative DOC changes are provided in the attachment to this response and affect the following DOCs: ITS 3.3.1.1 DOC A.16, ITS 3.3.2.1 DOC A.4, ITS 3.3.4.1 DOC A.6, ITS 3.3.5.1 DOC A.8, ITS 3.3.5.2 DOC A.8, ITS 3.3.6.1 DOC A.13, ITS 3.3.6.3 DOC A.7, and ITS 3.3.8.1 DOC A.6. Furthermore, ITS 3.3.8.2 DOC A.5 inadvertently used the term Allowable Value and has been deleted. In addition, the DOCs that describe the actual changes to the CTS Table values for those BSI items listed above have been modified to clearly state that the change in the CTS Table value reflects a change to the OPERABILITY value, not a change to the actual Allowable Value. The proposed changes to these DOCs are also provided in the attachment to this response and affect the following DOCs: ITS 3.3.1.1 DOC L.12, ITS 3.3.2.1 DOC L.6, ITS 3.3.4.1 DOC L.4, and ITS 3.3.6.1 DOC L.9.

Date Created: 03/16/2006 01:18 PM by David Roth
Last Modified: 03/28/2006 07:13 AM

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NRC ITS TRACKING

NRC Reviewer

ID	200603221611 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 ITS Number: 3.3.1.1 DOC Number: M.14 JFD Number: None Bases JFD Number: None Page Number(s): 27
Comment	Comment to document e-mail/phone discussion. No additional reply needed. Dave: I asked site engineering to calculate the following, so you can compare the two frequencies: Monticello Full Power = 1775 MW Monticello Core Load = 92.11 STU 2000 EFPH @ 1775 MW (Full Power) = 2000 EFPH / (24 H/Day) = 83.3 days 1000 MWD/STU @ 92.11 STU and 1775 MW (Full Power) = (92.11 STU * 1000 MWD/STU)/(1775 MW) = 51.9 days Roger Scott -----Original Message----- From: David Roth [mailto:DER@nrc.gov] Sent: Wednesday, March 22, 2006 8:47 AM To: Scott, Roger Subject: EFP-Hrs vs. MWD/T Rogeer: A real quick question. I'm trying to see how similar the frequencies are for the SRs below. I need to know how the 1000 MWD/T relates to 2000 EFP-hrs. Can you please e-mail the answer to me this morning? Standard: Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.6). New: Changes in APRM neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 2000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.6). Plant Justification: The Frequency for ISTS SR 3.3.1.1.6 has been changed from 1000 MWD/T to 2000 effective full power hours. This Surveillance Frequency is consistent with current plant practice for calibration of the local power range monitors. Rated Thermal Power - Rated thermal power means a steady-state power level of 1775 thermal megawatts. David E. Roth Reactor Systems Engineer Technical Specifications Branch

	NRR/ADRO/DIRS/ITSB **MS O12-H2** **U.S. Nuclear Regulatory Commission** **Washington, DC 20555** **(301) 415-2749 (work)** **(301) 415-3031 (fax)**
Issue Date	**03/22/2006**

Close Date	**03/22/2006**	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 03/27/2006	Monticello acknowledges that no additional reply is necessary.

Date Created: 03/22/2006 04:11 PM by David Roth
Last Modified: 03/22/2006 04:11 PM

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NRC ITS TRACKING

NRC Reviewer

ID	**200603271003** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	**ITS Section:** 3.3 **ITS Number:** 3.3.1.1 **DOC Number:** A.8 **JFD Number:** None **Bases JFD Number:** None **Page Number(s):** 3
Comment	**"The licensee also proposes to remove the time 'within 8 hours' for Required Condition A so that it reads, 'All operable control rods fully inserted.' As a result, control rods would be required to be inserted in accordance with TS 3.1.B.3, allowing a time frame defined by normal operating procedures, instead of within 8 hours. " ITS states to be in Mode 3 in 12 hours. Please clarify if A.8 should be M instead of A, since the mode switch position has been added. A ADMINISTRATIVE CHANGES — Changes to the CTS that do not result in new requirements or change operational restrictions or flexibility. M MORE RESTRICTIVE CHANGES — Changes to the CTS that result in added restrictions or reduced flexibility. ******** Clarification Below: Quoted first part of comment is from ML020920319, which was the last amendment which addressed what the "All operable control rods fully insterted" means. The citation was inadvertantly left off. It is mentioned to emphasize that control rods would be inserted in accordance with normal operating procedures. Current DOC A.8 states, "Therefore since the end result is equivalent [meaning all rods in], this change is acceptable." Please confirm that normal operating procedures under the CTS always required mode change to SHUTDOWN as well as all rods in. Please verify that CTS would not permit mode switch in STARTUP with all rods inserted to satisfy the CTS given a Required Condition A event, since this end result would not be equivalent to the proposed ITS.**
Issue Date	**03/27/2006**

Close Date	**03/28/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 03/27/2006	When the requirements of CTS 3.1.B are not met for the Mode Switch in Shutdown, Manual Scram, Neutron Flux IRM High - High, Neutron Flux IRM Inoperable, High Reactor Pressure, High Drywell Pressure, Reactor Low Water Level, and Scram Discharge Volume High Level (East and West) Trip Functions (CTS Table 3.1.1 Trip Functions 1, 2, 3.a, 3.b, 5, 6, 7, 8.a, and 8.b), CTS Table

	3.1.1 (Required Condition A) requires all OPERABLE control rods to be fully inserted. The purpose of the CTS Table 3.1.1 Required Condition A is to place the plant in a condition where RPS Instrumentation is not required to be OPERABLE; i.e., to exit the Applicability of the affected Functions. As shown in CTS Table 3.1.1, the Applicability for these Functions include either Run and Startup, or Startup only. Thus, inserting all OPERABLE control rods for this Required Condition includes placing the reactor mode switch in Shutdown, since this is required to exit the Applicability for these Functions. Furthermore, CTS 3.1.B.3 states that the plant must be placed in the required specified conditions (i.e., one of the Required Conditions referenced in CTS Table 3.1.1) "using normal operating procedures." The Monticello shutdown procedure requires the reactor mode switch to be placed in the shutdown position after all the control rods are inserted, if shutting down the plant by individually inserting each control rod. The other method specified in the Monticello shutdown procedure for shutting down the plant includes placing the reactor mode switch in the shutdown position while still critical. Therefore, the normal operating procedures require the reactor mode switch to be placed in the shutdown position. Therefore, Monticello considers this change strictly as an administrative clarification; it is not a more restrictive change. In addition, the NRC's question quoted a statement from a previous license amendment request concerning the time allowed to insert the control rods (i.e., 8 hours). This time limit was removed as part of License Amendment 103, with the only requirement that the control rods be inserted using normal operating procedures.

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NRC ITS TRACKING

NRC Reviewer

ID	200603290815 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3; ITS Number: 3.3.1.1; DOC Number: A.16; JFD Number: None; Bases JFD Number: None; Page Number(s): 7
Comment	DOC A.16 stated the terms "setpoint" and "Limiting Trip Settings" in the CTS is the same as the use of the term "Allowable Value" in the ITS. For each BSI re-classified as CLB in the Table 3.1.1, please provide diagrams of CTS Limiting Trip Settings and ITS Allowable Values. The diagram should be able to be used to show how the two different "Limiting Trip Settings" and "Allowable Values" (in CTS and ITS) came from the same existing calculation. Please also provide a diagram of Reactor Vessel Steam Dome Pressure CTS Limiting Trip Setting and ITS Allowable Value. Please also provide similar diagram for Reactor Low Water Level. The diagrams should show the relationship of the "Limiting Trip Settings" in CTS Table 3.1.1 to the values specified in ITS. For each of the non-reclassified items in CTS Table 3.1.1, please, on a per-item-# basis, re-state if the listed CTS value is the same number from the same calculation as the equivalent ITS "Allowable Value." For Item#11, it is recognized that CTS did not list a value.
Issue Date	03/29/2006

Close Date	03/31/2006	Resolution requires change to: None Docket Response Required? Yes

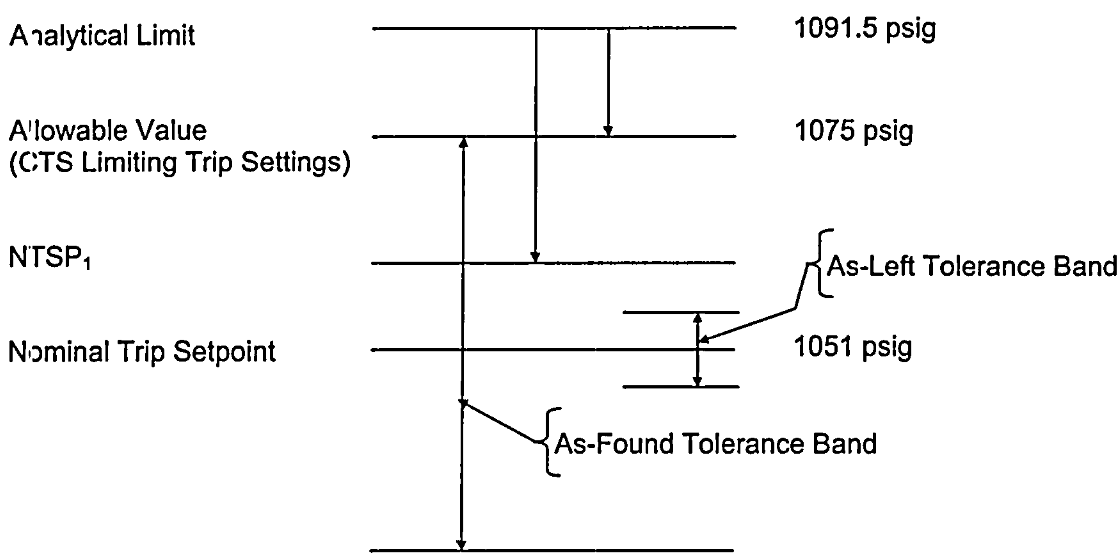
Responses

Licensee Response by Jerry Jones on 03/30/2006	The changes to the Current Technical Specifications (CTS) values for CTS Table 3.1.1 Trip Function 3.a (Neutron Flux IRM High-High), Trip Function 4.a (Flow Referenced Neutron Flux APRM High-High), and Trip Function 4.c (Flow Referenced Neutron Flux APRM High Flow Clamp) were reclassified as CLB in the 3/12/06 NMC response to RAI 200510281243. A diagram is attached to this response for each of these Trip Functions showing the relationship between the CTS "Limiting Trip Settings" and the ITS "Allowable Values." Also attached are similar diagrams for Trip Function 5 (High Reactor Pressure) and Trip Function 7 (Reactor Low Water Level). For CTS Table 3.1.1 Trip Function 6 (High Drywell Pressure), Trip Function 8 (Scram Discharge Volume High Level), Trip Function

	10 (Main Steamline Isolation Valve Closure), and Trip Function 12 (Turbine Stop Valve Closure), the associated calculations indicate that the CTS "Limiting Trip Setting" is the same value as the ITS "Allowable Value."

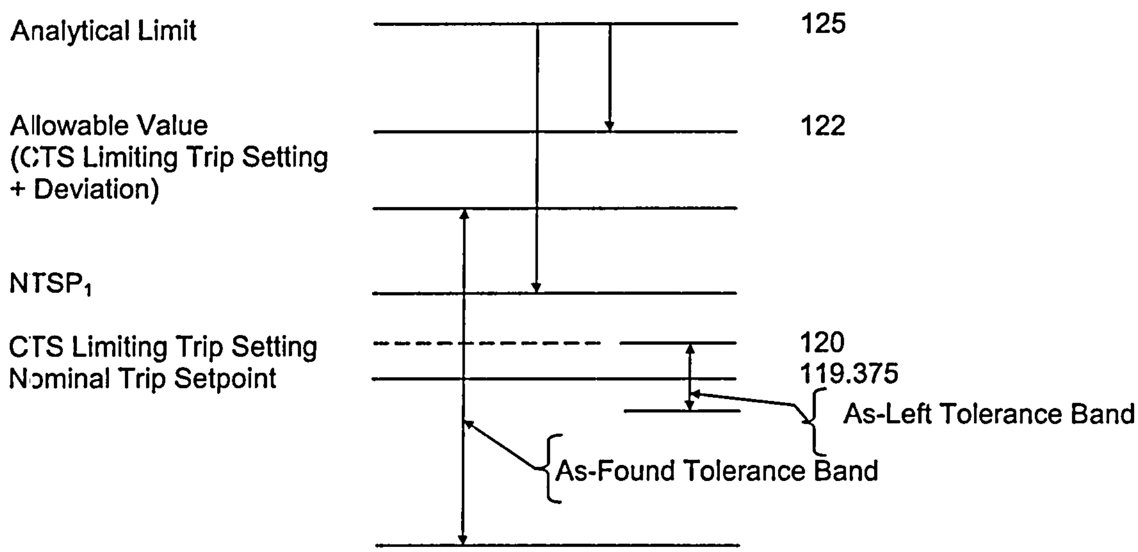
Date Created: 03/29/2006 08:15 AM by David Roth
Last Modified: 03/31/2006 10:06 AM

CTS Table 3.1.1 Function 5, Reactor Vessel Steam Dome Pressure - High



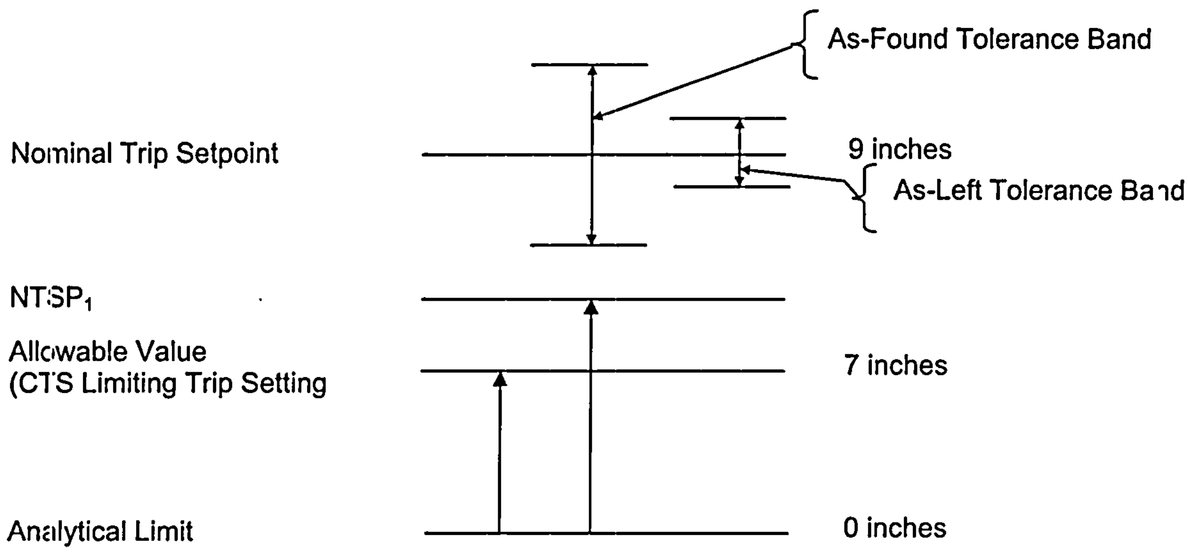
$NTSP_1$ is the term most similar to LSSS on the NRC diagram since it is the most limiting value of the setpoint that protects the AL.

CTS Table 3.1.1 Function 3.a, IRM Neutron Flux High - High



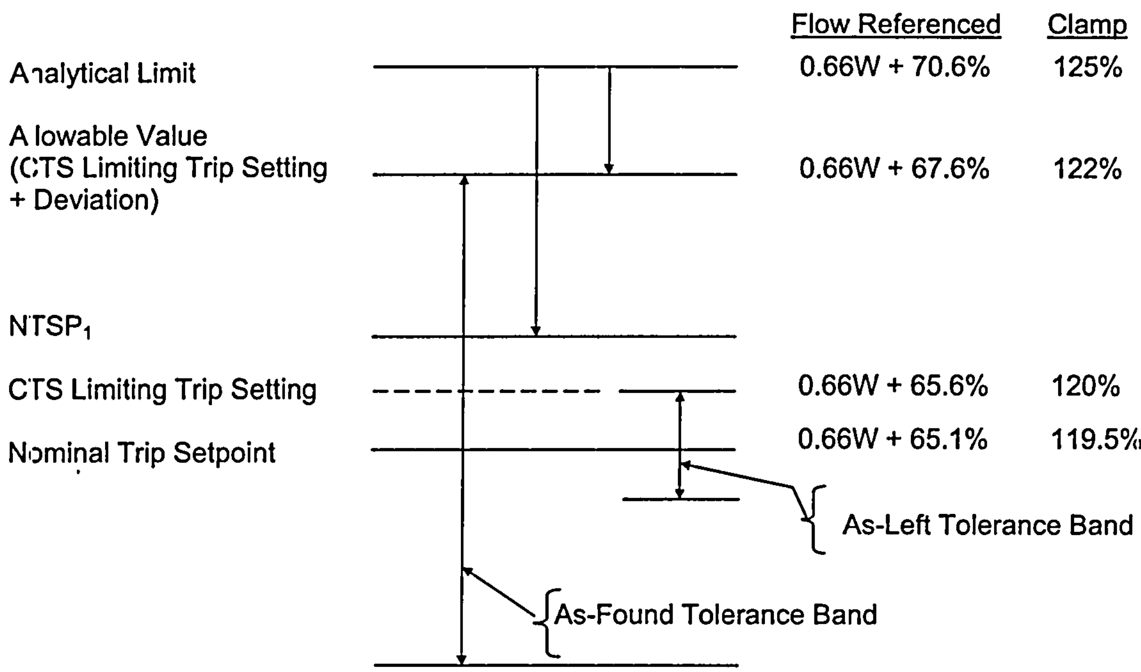
NTSP$_1$ is the term most similar to LSSS on the NRC diagram since it is the most limiting value of the setpoint that protects the AL.

CTS Table 3.1.1 Function 7, Reactor Vessel Water Level - Low



$NTSP_1$ is the term most similar to LSSS on the NRC diagram since it is the most limiting value of the setpoint that protects the AL.

CTS Table 3.1.1 Function 4.a and 4.c, APRM Flow Referenced Neutron Flux-High and Clamp



Clamp values are for flows (W) above 82.4%.

NTSP$_1$ is the term most similar to LSSS on the NRC diagram since it is the most limiting value of the setpoint that protects the AL.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200604061204 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: None; DOC Number: None; JFD Number: None; Page Number(s): ; ITS Number: None; Bases JFD Number: None
Comment	Please validate the Change Type classifications for ITS 3.3.1.2 LA.1 (looks like 3) ITS 3.5.1 LA. 4 (looks like a 6) ITS 3.6.1.2 LA. 1 (looks like a 3)
Issue Date	04/06/2006

Close Date	04/06/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 04/06/2006	1. ITS 3.3.1.2 LA.1: This Discussion of Change (DOC) justifies moving the CTS 3.10.B.1 requirement (Attachment 1, Volume 8, Rev. 0, Pages 114 and 121 of 760) that the SRMs be inserted to the normal operating level to the Bases. The Monticello ITS submittal classified this change as a type 2 change, which is removing descriptions of system operation. A type 3 change is removing procedural details for meeting TS requirements or reporting requirements. The NRC classified this change as a type 2 change as shown in the NRC Safety Evaluation Report for the last BWR ITS conversion (FitzPatrick Nuclear Power Plant, Amendment 274, dated 7/3/02). Therefore, Monticello believes that a type 2 classification is acceptable. 2. ITS 3.5.1 LA.4: This DOC justifies moving the CTS 4.5.A.5 Core Spray delta p instrumentation surveillances (Attachment 1, Volume 10, Rev. 0, Pages 7 and 16 of 118) to the Technical Requirements Manual (TRM). The Monticello ITS submittal classified this change as a type 4 change, which is removing performance requirements for indication-only instrumentation and alarms. A type 6 change is the removal of LCO, SR, or other TS requirement to the TRM, USAR, ODCM, OQAP, IST Program, or IIP. While it could be classified as a type 6 change, a type 4 change is a small subset of a type 6 change, thus is more accurate. Therefore, Monticello believes that a type 4 classification is acceptable. 3. ITS 3.6.1.2 LA.1: This DOC justifies moving the CTS 3.7.A.2.c information of what constitutes an OPERABLE air lock door (Attachment 1, Volume 11, Rev. 0, Pages 29 and 32 of 431) to the Bases. The Monticello ITS submittal classified this change as a type 1 change, which is removing details of system design and system description. A type 3 change is

	removing procedural details for meeting TS requirements or reporting requirements. The design of the primary containment air lock doors is for the doors to remain closed at all times, except when being used for entry and exit, and then only one of the doors is allowed to be open. This is ensured by the air lock interlock mechanism, which is designed to only allow one door at a time to be open. Therefore, Monticello believes that a type 1 classification is acceptable.

Date Created: 04/06/2006 12:04 PM by David Roth
Last Modified: 04/06/2006 04:57 PM

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NRC ITS TRACKING

NRC Reviewer

ID	200604070903 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: None ITS Number: None DOC Number: None JFD Number: None Bases JFD Number: None Page Number(s):
Comment	**Please validate or clarify to support change category of Admin. 3.6.2.1 A.3 The DOC state "The logging requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance records to maintain records of activities affecting quality, including the results of tests (i.e., Technical Specification Surveillances)." But the CFR does not explicity state to log items every five minutes. Will admin controls require logging every five mintues? 3.10.6 A2 (suggest adding "CTS Bases 3.10.E states that when the refueling interlock input signal from a withdrawn control rod is bypassed, administrative controls will be in effect to prohibit fuel from being loaded into that control cell.")**
Issue Date	04/07/2006

Close Date	04/10/2006 Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 04/07/2006	1. ITS 3.6.2.1. Discussion of Change (DOC) A.3: SR 3.6.2.1.1 (Attachment 1, Volume 11, Rev. 0, Page 232 of 431) requires verification that the suppression pool temperature is less than or equal to 100 degrees F every 5 minutes when performing testing that adds heat to the suppression pool. DOC A.3 (Page 222 of 431) states that the logging requirement of CTS 4.7.A.1.b (Page 220 of 431) is not needed to be specified in the ITS since 10 CFR 50 Appendix B, Section XVII duplicates the logging requirement. While the CFR does not specifically state to "log" this item every 5 minutes, since ITS SR 3.6.2.1.1 is required to be performed every 5 minutes, then it must be recorded (i.e., logged) to comply with the CFR requirement. The applicable Monticello procedure will continue to require the value to be logged every 5 minutes to ensure compliance with the ITS SR 3.6.2.1.1 requirement and 10 CFR 50 Appendix B, Section XVII. 2. ITS 3.10.6 DOC A.2: DOC A.2 (Attachment 1, Volume 15, Rev. 0, Page 107 of 178) justifies adding the ITS 3.10.6.b requirement that all other control rods in core cells containing one or more fuel assemblies are fully inserted. The CTS 3.10.E Bases statement referred to in the NRC RAI is discussing administrative controls to ensure fuel is not inserted into a core cell when the control rod for that cell has

	been removed. This Bases statement does not affect the ITS 3.10.6.b requirement, therefore Monticello does not believe that the statement is needed in DOC A.2.

Date Created: 04/07/2006 09:03 AM by David Roth
Last Modified: 04/10/2006 11:42 AM

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200604070922** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	**ITS Section:** None **DOC Number:** None **JFD Number:** None **Page Number(s):** **ITS Number:** None **Bases JFD Number:** None
Comment	**Please clarify DOC 3.0 A.3 to show better how ITS LCO 3.0.6 is consistent with Monticello's current practices and therefore administrative.**
Issue Date	**04/07/2006**

Close Date	**04/10/2006** Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 04/07/2006	Discussion of Change A.3 (Attachment 1, Volume 5, Rev. 0, Pages 12 and 13 of 63) provides the justification for why the addition of ITS LCO 3.0.6 is administrative. This DOC is consistent with the DOC provided in the last two NRC-approved ITS conversions (DC Cook Nuclear Plant Units 1 and 2, Amendments 287 and 269, dated 6/1/05, and FitzPatrick Nuclear Power Plant, Amendment 274, dated 7/3/02). Therefore, Monticello does not believe that any additional information is necessary.

Date Created: 04/07/2006 09:22 AM by David Roth
Last Modified: 04/10/2006 11:40 AM

Close

NRC ITS TRACKING

NRC Reviewer

ID	**200604111049** Conference Call Requested? **No**
Category	**Discussion**
ITS Information	ITS Section: **None** ITS Number: **None** DOC Number: **None** JFD Number: **None** Bases JFD Number: **None** Page Number(s):
Comment	**Please advise regarding current NMC and QATR nomenclature.**
Issue Date	**04/11/2006**

Close Date	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 04/13/2006	CTS 6.1.B.1 (Volume 1, Attachment 17, Rev. 0, Page 15 of 143) references the Operational Quality Assurance Plan (OQAP), which was the Monticello 10 CFR 50 Appendix B program description. However, NMC submitted a common Quality Assurance Topical Report (QATR) on 10/31/03 to replace the existing OQAP at the NMC sites, including Monticello. The NRC approved the NMC QATR in the safety evaluations dated 1/13/05 and 3/24/05. However, a concurrent Technical Specification license amendment request to change the existing title from OQAP to QATR was not made. Therefore, the ITS submittal is being changed to include the proper QA program plan title (QATR). The affected ITS submittal pages are included in the attachment to this response. Furthermore, it was noted that Discussion of Change (DOC) LA type 6 includes the term "OQAP" in the title. However, no LA type 6 changes were made that affects the QA program description. Therefore, in lieu of changing all the LA type 6 DOCs to include the new title (i.e., QATR), the old title (i.e., OQAP) will remain in the LA type 6 title (Note: the title is in the parentheses at the beginning of each LA DOC).

Date Created: 04/11/2006 10:49 AM by David Roth
Last Modified: 04/11/2006 10:49 AM

Assign | New Response | Close

NRC ITS TRACKING

NRC Reviewer

ID	200604131047 Conference Call Requested? No
Category	Minor Technical
ITS Information	ITS Section: 3.3 DOC Number: L.14 JFD Number: None Page Number(s): 27 ITS Number: BSI 2 Bases JFD Number: None
NRC Owner	Terry Beltz
Comment	During your discussion of change for the testing of the Manual Scram Function, you state: "ITS SR 3.3.1.1.4 has been included to ensure the automatic logic scram relays are tested every week. A review of past Manual Scram functional test Surveillances was performed and all completed tests were successful. Both monthly and weekly tests performed in 1992 (pre- and postimplementation of the monthly to weekly Surveillance Frequency change) and recent weekly tests were reviewed. In total, 27 completed Surveillances were reviewed and the Manual Scram functional test was successful in every case." Please clarify: 1) The scope of your review of past Manual Scram function test Surveillances. 2) Why were there only 27 completed Surveillances reviewed, when there have been approximately 156 weeks since 1992. What was the extent of this review?
Issue Date	04/13/2006

Close Date	Resolution requires change to: None Docket Response Required? No

Responses

Licensee Response by Jerry Jones on 04/17/2006	Two types of reviews were performed for the Manual Scram Functional Test requirement. A review of the Monticello Corrective Action Program was performed to determine failures of the procedure used to perform the Manual Scram Functional Test. The results of the Monticello Corrective Action Program review revealed no Manual Scram Functional Test failures. The 27 tests reviewed consisted of: 11 monthly tests performed immediately prior to implementing the weekly test frequency associated with Amendment 81; three weekly tests performed immediately following implementation of Amendment 81; and 13 weekly tests recently performed at the time this information was evaluated for extension of the surveillance frequency from 7 days to 31 days. The results of this review revealed no Manual Scram Functional Test failures.

Close

NRC ITS TRACKING

NRC Reviewer

ID	200604171314 Conference Call Requested? No
Category	Discussion
ITS Information	ITS Section: 3.3 DOC Number: A.16 JFD Number: None Page Number(s): ITS Number: 3.1.1 Bases JFD Number: None
Comment	Monticello's DOC A.16 (page 21 of 760) was modified via RAI 200603161318, but appears to need additional clarification to be consistent with 200603290815. In the draft SE provided to Monticello, the table entry for DOC A.16 was highlighted and marked as needing clarification from the licensee. The modified version states, in part, "Therefore, the use of the terms 'setpoint' and 'Limiting Trip Settings' in the CTS is the same as the use of the term 'Allowable Value' in the ITS." The above sentence appears to be inconsistent with the answer provided to 200603290815. The diagrams provided with the 200603290815 showed that, in some instances (e.g., IRM High-High) the CTS Limiting Trip Setting was not the same as the ITS AV, while in other cases (e.g., Rx Wtr Lvl), they were the same. In RAI 200603290815, the licensee stated, The changes to the Current Technical Specifications (CTS) values for CTS Table 3.1.1 Trip Function 3.a (Neutron Flux IRM High-High), Trip Function 4.a (Flow Referenced Neutron Flux APRM High-High), and Trip Function 4.c (Flow Referenced Neutron Flux APRM High Flow Clamp) were reclassified as CLB in the 3/12/06 NMC response to RAI 200510281243. A diagram is attached to this response for each of these Trip Functions showing the relationship between the CTS "Limiting Trip Settings" and the ITS "Allowable Values." Also attached are similar diagrams for Trip Function 5 (High Reactor Pressure) and Trip Function 7 (Reactor Low Water Level). For CTS Table 3.1.1 Trip Function 6 (High Drywell Pressure), Trip Function 8 (Scram Discharge Volume High Level), Trip Function 10 (Main Steamline Isolation Valve Closure), and Trip Function 12 (Turbine Stop Valve Closure), the associated calculations indicate that the CTS "Limiting Trip Setting" is the same value as the ITS "Allowable Value." The diagram provided for Neutron Flux IRM High-High shows the CTS Limiting Trip setting to be different than the ITS Allowable Value. The diagram provided for APRM Flow Ref Clamp shows the CTS Limiting Trip settings to be different than the ITS Allowable Values. The diagram provided for High Rx Pressure shows the AV and the CTS Limiting Trip Setting to be the same. The diagram provided for Rx Water Lvl shows the AV and the CTS Limiting Trip Setting to be the same. QUESTION: WILL THE DOC A.16 BE REPLACED WITH THE ANSWERS FROM 200603290815?
Issue Date	04/17/2006

Close Date	04/20/2006	Resolution requires change to: None Docket Response Required? **No**

Responses

Licensee Response by Jerry Jones on 04/19/2006	NMC will modify ITS 3.3.1.1 Discussion of Change (DOC) A.16 (Attachment 1, Volume 8, Page 21 of 760), as shown in the attached response. The revised DOC A.16 more clearly states that the change the DOC is justifying is an administrative change. This revised DOC A.16 supersedes the change provided in the NMC response to RAI 200603161318.

Date Created: 04/17/2006 01:14 PM by Terry Beltz
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