FENOC
FirstEnergy Nuclear Operating Company

5501 North State Route 2
Oak Harbor, Ohio 43449

Barry S. Allen
Vice President - Nuclear

419-321-7676
Fax: 419-321-7582

September 3, 2008

L-08-224

10 CFR 50.90

ATTN: Document Control Desk
United States Nuclear Regulatory Commission
Washington, D. C. 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit 1
Docket No. 50-346, License No. NPF-3
Response to Request for Additional Information Regarding License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS), and Copy of Two Questions from the U.S. Nuclear Regulatory Commission and Davis-Besse Nuclear Power Station Improved Technical Specifications Conversion Website (TAC No. MD6398)

By letter dated August 3, 2007, as supplemented by letters dated October 29, 2007, May 16, 2008 (2 letters), July 23, 2008, August 07, 2008 and August 26, 2008, FirstEnergy Nuclear Operating Company (FENOC) submitted an application to amend the Technical Specifications of Davis-Besse Nuclear Power Station, Unit 1 (DBNPS), revising the current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications (ISTS) as described in NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Revision 3.1, and certain generic changes to the NUREG.

By letter dated August 26, 2008, FENOC provided a copy of applicable portions of the U.S. NRC and DBNPS ITS Conversion Website. The August 26, 2008 letter did not include copies of two questions (200712171517 and 200802121126). FENOC's response to these questions is provided in Attachment 1, suitable for posting on the DBNPS docket. The files attached to the 7/01/2008 Licensee website response to question 200712171517 and to the 8/15/2008 Licensee website response to question 200802121126 are not included in Attachment 1, since the information in those files is addressed within this letter.

A001
NRR

By letter dated June 18, 2008, the Nuclear Regulatory Commission (NRC) staff requested additional information necessary to complete the amendment application review. The FENOC responses to the NRC staff's questions contained in the information request are provided in Attachment 2.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

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

Executed on September 3, 2008

Sincerely,

Barry S. Allen

Barry S. Allen, Vice President-Nuclear

Attachments:

1. Copy of questions 200712171517 and 200802121126 from the U.S. Nuclear Regulatory Commission (NRC) and Davis-Besse Nuclear Power Station (DBNPS) Improved Technical Specification (ITS) Conversion Website.

2. Response to Request for Additional Information Regarding License Amendment Request: Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS)

cc: (all w/o Attachment)
NRC Region III Administrator
NRR Project Manager
NRC Resident Inspector
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Attachment 1
L-08-224

Copy of questions 200712171517 and 200802121126 from the U.S. Nuclear Regulatory Commission (NRC) and Davis-Besse Nuclear Power Station (DBNPS) Improved Technical Specification (ITS) Conversion Website

Section 3.8 RAIs

Question 200712171517

Return to View Menu | Print Document

RAI Screening Required: Yes

Status: Closed

This Document will be approved by: **George Wilson**

Regulatory Basis must be included in Comments section of this Form

This document has been reviewed and information in this question contains NO SUNSI sensitive material (the checkbox to the right must be selected before this question can be submitted)

Yes

NRC ITS TRACKING

NRC Reviewer

ID	**200712171517**		Conference Call Requested? **No**	
Category	**BSI - Beyond Scope Issue**			
ITS Information	ITS Section: **3.8 Robert Clark**	TB POC: **Vijay Goel**	JFD Number: **None**	Page Number(s): **20**
	ITS Number: **3.8.1**	OSI: **None**	DOC Number: **M.6**	Bases JFD Number: **None**
Comment	**Regulatory Bases: 10CFR50.36(c)(3), GDC 18 "Based on history of recent long duration of Loss-of-Offsite Power (LOOP), the mission time of EDG is required to be minimum 24 hours. Therefore, NRC staff recommends a 24 hours endurance run of EDG under ITS (SR 3.8.1.13), which is also in accordance with RG 1.9. Therefore, either 24 hours endurance run should be accepted or provide justification that the reliability of the EDG after 8 hours of run will not be impacted." "Provide the worst case EDG load profile for 24 hours indicating both the KW and PF values for each segment of loading for a design basis accident."**			
Issue Date	**12/17/2007**			
Close Date	**08/11/2008**			

Logged in User: Anonymous

Responses

Licensee Response by Jerry Jones on 02/11/2008	Davis-Besse currently does not have a Surveillance Requirement to perform an EDG endurance run. Davis-Besse reviewed other, recently-approved ITS submittals and determined that the last three NRC-approved ITS submittals only required an 8 hour EDG endurance run, not a 24 hour run. Thus, the NRC has approved exceptions to the 24 hour run, and allowed it to be 8 hours as specified in IEEE-387-1995. The three recently-approved ITS submittals are DC Cook Units 1 and 2 (Amendments 287 and 269, approved June 1, 2005, ADAMS Accession No. ML050620034), Monticello Nuclear Generating Plant (Amendment 146, approved June 5, 2006, ADAMS Accession No. ML061240241), and Beaver Valley Power Station Units 1 and 2 (Amendments 278 and 161, approved February 21, 2007, ADAMS Accession No. ML070160593). The Davis-Besse Justification for Deviation (JFD) 16 provided to justify the change from 24 hours to 8 hours (Volume 14, Page 62) is consistent with the JFDs provided by Monticello and Beaver Valley (DC Cook's JFD stated it was consistent with current licensing basis). Therefore, since the NRC has approved an 8-hour endurance run for the last

	three approved ITS submittals, Davis-Besse believes that this precedent should be applicable to the Davis-Besse ITS submittal and an 8-hour endurance run approved for Davis-Besse. In addition, the response to question 200710091446 provides information concerning the maximum accident loading values for the EDGs.
NRC Response by Vijay Goel on 03/10/2008	In the past and also recently, there have been a number of reported emergency diesel generator (EDG) failures after 8 hours of the 24 hours endurance run at other nuclear plants. Therefore, the staff considers it necessary that an EDG endurance run should be for 24 hours to provide confidence that it will run successfully during the accident scenarios to complete its mission. The licensee should reconsider revising the proposed endurance run from 8 hours to 24 hours in the proposed ITS SR 3.8.1.13.
Licensee Response by Bill Bentley on 04/04/2008	On a 3/19/08 phone call, it was stated that Davis-Besse staff would be consulted to reconsider the 8 hour duration. After consulting with Davis-Besse staff, Davis-Besse does not desire to make any changes to the ITS submittal to address this question.
NRC Response by Vijay Goel on 05/05/2008	For the purpose of verification of EDG loading values provided in the response to questions 200710091446 and 200712171517, Davis-Besse provided (on 4/29/08) an excerpt from the AC Power System Analysis calculation. The NRC staff has the following questions on the EDG loading results. In addition, the staff needs a clarification on ITS SR 3.8.1.13, Note 3, as follows: 1. The EDG 1-1 Loading Results (page 84 of the calc): Corresponding to LOOP/LOCA (ETAP case LF7e), the ETAP loading is identified as 2322 kW which matches with the licensee's response to question 200710091446 and 200712171517. However, the total KW loading after adding 122 kW based on Section 4.2.3.1 is 2444 kW. Since Section 4.2.3.1 has not been included in the calc excerpt, we are not sure what 122 kW represents. Regarding the 122 kW load: Clarify why it is not included in the kW loading provided in response to questions 200710091446 and 200712171517. 2. The EDG 1-2 Loading Results (page 86 of the calc): Corresponding to LOOP/LOCA (ETAP case LF10e), the ETAP loading is identified as 2384 kW which matches with the licensee's response to question 200710091446 and 200712171517. However, the total KW loading after adding 123 kW based on Section 4.2.3.2 is 2507 kW. Since Section 4.2.3.2 has not been included in the calc excerpt, we are not sure what 123 kW represents. Regarding the 123 kW load: Clarify why it is not included in the kW loading provided in response to questions 200710091446 and 200712171517. 3. The EDG 1-1 Loading Results (page 84 and 85 of the calc): Corresponding to the Appendix R loading scenario (ETAP case LF9), the total kW loading is 2627 kW, which is approximately 101% of the continuous rating of the EDG (which is 2600 kW). The staff position has been that EDG testing should envelop the worst case design bases loading scenario (e.g., Dresden TIA). Therefore, Davis-Besse needs to test the EDG at 101% or higher of the continuous rating of EDG (after 2 hour test at 105% to 110% rating). Therefore, 90% to 100% loading presently proposed in the ITS is not acceptable, unless adequately justified. Justify how EDG testing to 90% to 100% loading of the continuous rating (presently proposed in the ITS SR 3.8.1.13) envelops the worst case Appendix R loading scenario. ITS SR 3.8.1.13, Note 3: In this note, it is stated that: "If part b is performed with EDG synchronized with the offsite power, it shall be performed within the power factor limit..." Clarify why the words "at a power factor" in the STS has been modified to "within the power factor" in the ITS.
Licensee Response by Bill Bentley on 05/09/2008	Questions 1 and 2 responses In response to the NRC questions regarding 200710091446 and 200712171517 for clarification on ITS 3.8.1.13, specifically regarding the Emergency Diesel Generator (EDG) loading results that were provided to the NRC (on 4/29/08): Sections 4.2.3.1 and 4.2.3.2 of Calculation C-

EE-015.03-008, R.4, calculate the additional load (in kW) on the Emergency Diesel Generators (EDGs) due to the "maximum frequency effect". The EDGs have a maximum frequency of 61.2 Hz (102% of rated 60 Hz per section 1.11.27 of the calculation). This maximum frequency value in the calculation was conservatively chosen so that it bounds the existing design basis value. The additional load due to frequency variation is added to the calculated loading results shown in sections 5.9.3 and 5.10.3 of calculation C-EE-015.03-008, R.04, to calculate the expected kW loading of the EDGs while operating at a frequency of 61.2Hz. The expected kW loading is then verified to have kilowatt (kW) and kilovolt-ampere (kVAR) loading at values less than the criteria established in section 1.11.26 to C-EE-015.03-008, R.04. The original response to the questions regarding 200710091446 and 200712171517 did not include the effects of frequency variation in the total EDG load value. The values that were provided were for nominal frequency (60Hz) loading. Accounting for the additional load due to the assumed maximum frequency, the maximum expected, automatically connected load on EDG 1-1 is 2444kW and the maximum expected, automatically connected load on EDG 1-2 is 2507kW (per calculation C-EE-015.03-008, R.04). The endurance test and load test, ITS SR 3.8.1.13 (Volume 13, Page 45), require the EDGs to be operated for greater than or equal to 8 hours, with greater than or equal to 2 hours loaded at greater than or equal to 2730kW and less than or equal to 2860kW. For the remaining 6 hours of the test, the EDGs are loaded at greater than or equal to 2340kW and less than or equal to 2600kW. The loading applied during the endurance test for the first 2 hours will continue to bound the maximum automatically connected load calculated in C-EE-015.03-008, R.04, regardless of the impact of EDG loading due to the frequency variation. Furthermore, the steady state frequency band for the ITS SRs is 59.5Hz to 60.5Hz, which is more limiting than that assumed in the EDG loading calculation. Thus, if the EDGs start and load at 60.5Hz following an accident, the 2 hour test duration at a minimum of 2730kW clearly demonstrates the capability of the EDG to handle this higher load, even at 61.2 Hz. In addition, the EDGs are procedurally monitored to be within 0.3Hz of their nominal frequency, 60Hz, prior to loading during EDG Monthly Surveillance Testing. The frequency value of 61.2Hz used in calculation C-EE-015.03-008, R.04, is conservatively assumed so that it bounds the frequency documented in the EDG Monthly Tests. This provides further assurance that the EDG loading determined in calculation C-EE-015.03-008, R.04, will remain bound by the endurance test load. Question 3 response: The NRC reviewer stated that the Appendix R loading scenario is required to be covered by the ISTS Surveillance. Davis-Besse does not believe that this is required to be covered. In the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 16, 1993, the NRC promulgated the NRC policy concerning the scope and purpose of Technical Specifications as required by 10 CFR 50.36. The four criteria for inclusion of requirements in Technical Specifications were provided in Section IV, The Commission Policy. Criteria 2 was specified as a process variable, design feature, or operating restriction that is an initial conditions of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Criterion 3 is a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The discussion for both these Criteria state that the analyses for the Design Basis Accident and Transients are those contained in Chapters 6 and 15 of the FSAR (or equivalent Chapters) and are identified as Condition II, III, or IV events (ANSI N 18.2). Furthermore, the Policy Statement states that the Commission will also consider the criteria in evaluating future generic

	requirements for inclusion in Technical Specifications. The Appendix R scenario is not a design basis accident or transient for Davis-Besse. Therefore, the loading requirements of proposed ITS SR 3.8.1.13 are based on the worst case design basis accident, which is the LOCA-LOOP scenario. The NRC reviewer also asked why the words in ISTS SR 3.8.1.14 Note 3 (Volume 13, Page 45) "at a power factor" had been changed to "within the power factor limit." Justification for Deviation (JFD) 14 (Pages 51 and 52) provide the reason why the words were changed. Essentially, the words were changed since the power factor limit is now in the Bases (as justified by JFD 14) and the words "within the power factor limit" are more consistent with similar words in other Surveillances. The ITS Bases identifies a different limit for each of the two EDGs. Thus, within the limit means that it can be at any value that is less than or equal to the limit specified in the ITS Bases for that given EDG.
Licensee Response by Bill Bentley on 07/01/2008	The NRC provided Davis-Besse with a formal written RAI, dated 6/18/2008. In that RAI, the NRC requested additional information related to EDG Appendix R loading. The RAI stated "Provide the loading profile for the Appendix R scenario to demonstrate that the proposed EDG endurance/margin test ensures that the analyzed functions can be performed. Alternately, confirm by calculation that the EDG Appendix R loading after 2 hours will be less than 100 percent of its continuous rating, considering the 60.5 hertz maximum steady state frequency proposed by the licensee in the ITS." The attached response from Design Engineering provides the requested information.
NRC Response by Vijay Goel on 07/17/2008	According to the ITS SR 3.8.1.13, Note 3, the licensee has proposed to take an exception to the STS for the "power factor limit" test for part "a", stating that: "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." The NRC staff does not agree with the above exception to the STS, since the purpose of "power factor limit" test for part "a" is also to demonstrate the margin in the KVAR capability of the machine (similar to demonstrating the margin in the KW capability of machine) during first two hours of the endurance run. Therefore, the ITS SR 3.8.1.13, Note 3 should make no distinction in the "power factor limit" test between part "a" or part "b" of the test.
Licensee Response by Jerry Jones on 07/22/2008	Davis-Besse does not believe that the first two hours of ITS SR 3.8.1.13 should be required to be performed at an accident analysis power factor limit. The first part of the test requires the EDG to be tested at 105% to 110% of the EDG rating. This is in excess of the post-accident loading value of the EDG. The 6 hour test values in ITS SR 3.8.1.13.b ensure the EDG is tested at loads equivalent to the normal post-accident loading value. This part of the test is required to be performed at the power factor limit. However, as stated in Justification for Deviation 14, ITS 3.8.1.13.a is testing the EDG at a load in excess of that assumed in the accident analysis. Since the accident analysis does not assume the EDGs are loaded beyond the normal band provided in ITS SR 3.8.1.13.b, there is no power factor limit for this part of the test. Davis-Besse has voluntarily added ITS SR 3.8.1.13 to perform the 8 hour load run, of which the 6 hours is being tested at the post-accident power factor limit (consistent with accident analysis assumptions). Davis-Besse does not desire to add a power factor limit to the overload portion of the test. Furthermore, Davis-Besse notes that the non-inclusion of a power factor limit for the overload portion of the test is consistent with the allowances approved in the Monticello ITS conversion (ML061070577), approved June 5, 2006 and also with the DC Cook Units 1 and 2 ITS conversion (ML050620034), approved June 1, 2005. (Note that for DC Cook, the endurance run was approved to be at the 90% to 100% load rating at the post-accident power factor limit for the entire duration. There was no overload load rating test required).

Licensee Response by Bill Bentley on 07/30/2008	Based on a recent NRC phone conversation, Davis-Besse will agree to add a power factor requirement to part a of ITS SR 3.8.1.13. The power factor limit for the 2-hour portion of the test will be 0.90, consistent with Regulatory Guide 1.9, Rev. 3. A draft markup regarding this change is attached. This change will be reflected in the supplement to this section of the ITS Conversion Amendment.
Licensee Response by Bill Bentley on 07/30/2008	Based on a recent NRC phone conversation, Davis-Besse will agree to add a power factor requirement to part a of ITS SR 3.8.1.13. The power factor limit for the 2-hour portion of the test will be 0.90, consistent with Regulatory Guide 1.9, Rev. 3. The first draft markup provided on 7/30/08 did not include all of the changes needed to JFD #14. A new draft markup regarding this change is attached and supercedes the previous 7/30/08 markup. This change will be reflected in the supplement to this section of the ITS Conversion Amendment.
NRC Response by Vijay Goel on 08/11/2008	No further questions at this time. Item closed.

Date Created: 12/17/2007 03:17 PM by Vijay Goel
Last Modified: 08/11/2008 04:45 PM

DC Cook information

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
4.8.1.1.2.e.7 SR 3.8.1.14 15	6
- NOTES - 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the ~~plant~~ unit is maintained or enhanced. INSERT 12 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~0.9~~ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.	5 13 4
Verify each DG ~~operating at a power factor ≤ [0.9]~~ operates for ≥ ~~24~~ hours 7 8 3150	10 4 ~~[18]~~ months 24 8
a. ~~For ≥ [2] hours loaded ≥ [5250] kW and ≤ [5800] kW and~~ 3500 at a	
b. ~~For the remaining hours of the test loaded ≥ [4500] kW and ≤ [5000] kW.~~	8

allowed to be performed at any time. The Note has not been added since the Surveillance test procedure currently allows the test to be performed during a shutdown of the unit from power operation (i.e., MODE 1) by tripping the main turbine and verifying the transfer to the preferred offsite circuit.

8. The DG endurance run time of ≥ 24 hours has been changed to ≥ 8 hours consistent with the current licensing basis as approved in License Amendment 207 (Unit 1) and 191 (Unit 2). The test is limited to the continuous rating consistent with the current licensing basis as approved in Licensing Amendment 125 (Unit 1) and 112 (Unit 2). The load range specified has been changed to values consistent with Regulatory Guide 1.9, Rev. 3, paragraph C.2.2.9 (90% to 100% of the continuous rating). The allowance to test the DG within the prescribed range is discussed in the Discussion of Changes for ITS 3.8.1.

9. The steady state limit does not apply to the simultaneous start of all DGs (ISTS SR 3.8.1.20), since it is a test of starting independence, not operating independence. This is consistent with the current licensing basis.

10. TSTF-276, Rev.2 was approved by the NRC on April 14, 2000. However, when NUREG-1431, Rev. 2 was issued, this TSTF was not completely included. Therefore, changes approved by TSTF-276, Rev. 2 have been made.

11. Editorial/grammatical error corrected.

12. ISTS SR 3.8.1.8 (ITS SR 3.8.1.9) has been revised to include two parts consisting of: a) a transfer from the auxiliary source (i.e., main generator) to the preferred offsite circuit; and b) a manual alignment to the alternate offsite circuit. These changes were made consistent with the current licensing basis. However, a Note has also been added to ISTS SR 3.8.1.8 (ITS SR 3.8.1.9) that states SR 3.8.1.9.a is only required to be met when the auxiliary source is supplying the electrical power distribution subsystem. This change is necessary since the automatic transfer from the auxiliary source to the preferred offsite circuit is not necessary when the preferred offsite circuit is supplying onsite power. In this situation the preferred offsite circuit is performing its function by supplying the onsite power.

13. TSTF-283, Rev. 3 was approved by the NRC on April 13, 2000. However, when NUREG-1431, Rev. 2 was issued, this TSTF was not completely included. Therefore, changes approved by TSTF-283, Rev. 3 have been made.

14. ISTS SR 3.8.1.12 part a has been modified by deleting the words "after auto-start and during tests" for consistency with similar words in ISTS SR 3.8.1.7 and SR 3.8.1.15. In addition, the words "auto-start" are redundant to the words in the first part of ISTS SR 3.8.1.12, and the words "and during tests" is not correct; the voltage and frequency limits of part b are different than those in part a of the SR.

15. ISTS 3.8.1 Required Actions A.1 and B.1 have been modified by the addition of Notes. The Note for Required Action A.1 states that the Required Action is not applicable if a require opposite unit offsite circuit is inoperable and the Note for Required Action B.1 states that the Required Action is not applicable if a required opposite unit DG is inoperable. With an opposite unit offsite circuit or DG

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	----------NOTE---------- This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each DG's automatic trips are bypassed on an actual or simulated loss of voltage signal on the emergency bus or an actual or simulated ESF actuation signal except: a. Engine overspeed; and b. Generator differential current.	24 months
SR 3.8.1.15	----------NOTES---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each DG operates for ≥ 8 hours at a load ≥ 3150 kW and ≤ 3500 kW.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	----------NOTE---------- This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. ---------- Verify each DG's automatic trips are bypassed on an actual or simulated loss of voltage signal on the emergency bus or an actual or simulated ESF actuation signal except: a. Engine overspeed; and b. Generator differential current.	24 months
SR 3.8.1.15	----------NOTES---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each DG operates for ≥ 8 hours at a load ≥ 3150 kW and ≤ 3500 kW.	24 months

Monticello information

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
DOC M.2 SR 3.8.1.14 (9) ----------NOTES---------- (9) 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with (E) DG synchronized with offsite power, it shall be performed at (part b is) a power factor ≤ [0.9] (limit). However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. (within the) (20) (3) (20) ---------- (10) (E) Verify each DG operates for ≥ 24 (8) hours: (2625)	[18] (24) months (3) (1)
a. For ≥ [2] hours loaded ≥ [3100] kW and ≤ [3400] (2750) kW and (;) (2)	(1)
b. For the remaining hours of the test loaded ≥ [2850] (2250) kW and ≤ [3150] (2500) kW.	(1)

JUSTIFICATION FOR DEVIATIONS
ITS 3.8.1, AC SOURCES - OPERATING

9. ISTS SR 3.8.1.13 has not been included in the Monticello ITS since it is not applicable to the Monticello EDG design. The Monticello EDG design does not include emergency diesel generator trips that are bypassed on a loss of voltage signal on the emergency bus concurrent with an ECCS initiation signal. Subsequent Surveillances have been renumbered, as applicable.

10. ISTS SR 3.8.1.14 requires each EDG to operate for ≥ 24 hours. ITS SR 3.8.1.9, which is a new requirement (see DOC M.2), requires each EDG to operate for ≥ 8 hours. The 8 hour duration for this test is considered sufficient to demonstrate EDG OPERABILITY. This change is based on the requirements of IEEE Standard 387-1995, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations." IEEE Standard 387-1995, Section 7.5.9 and Table 3 for the endurance and load test conducted during shutdown/refueling once every two years, state to demonstrate the load carrying capability for an interval of not less than 8 hours, of which 2 hours should be at a load equivalent to the short time rating of the diesel generator and 6 hours at a load equivalent to the 90%-100% of the continuous rating.

11. 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.

12. ISTS SR 3.8.1.18 requires verification that the interval between each sequenced load block is within ± 10% of design interval for each load sequence timer. The SR is proposed to be changed to not include the actual time limit and to delete the upper limit requirement, such that the interval between each load block is only required to be greater than or equal to the minimum design load interval.

    As stated in the ISTS Bases, the purposes of the 10% load sequence time interval tolerance are to ensure that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding emergency safeguards equipment time delays are not violated. The first purpose is met solely by applying a lower limit. If the interval between two load blocks is greater than minimum of the design interval, the capability of the EDG to perform its function is not necessarily impacted. For the first load interval, sufficient time after energizing the first load block to allow the EDG to restore frequency and voltage prior to energizing the second load block is still provided, since the minimum time needed is the minimum design interval; allowing more time than the design interval (e.g., plus 10%) does not negatively affect the ability of the EDG to perform its intended function, with respect to the first load interval. In addition, it is recognized that if there is an additional load block following the first two described above, then allowing the load interval between the first two load blocks to be longer than the minimum design interval could impact the capability of the EDG to restore frequency and voltage prior to the start of the third load block. However, the requirement that "each" load block be greater than or equal to the minimum design

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.9	----------NOTES---------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If part b is performed with EDG synchronized with offsite power, it shall be performed within the power factor limit. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ----------	
	Verify each EDG operates for ≥ 8 hours: a. For ≥ 2 hours loaded ≥ 2625 kW and ≤ 2750 kW; and b. For the remaining hours of the test loaded ≥ 2250 kW and ≤ 2500 kW.	24 months

Beaver Valley information

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~14~~ 10 (13)	
- NOTES - 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. (NUREG-1431, Rev. 3: Credit may be taken for unplanned events that satisfy this SR.) 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ ~~[0.9]~~ 0.89 (22). However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable.	
Verify each DG ~~operating at a power factor ≤ [0.9]~~ operates for ≥ ~~24~~ 8 hours: (NUREG-1431, Rev. 3)	[18] months
a. For ≥ ~~[2]~~ hours loaded ~~≥ [5250] kW and ≤ [5500] kW and~~ ≥ 2750 kW and ≤ 2850 kW (Unit 1) ≥ 4238 kW and ≤ 4535 kW (Unit 2), and (1) (14)	
b. For the remaining hours of the test loaded ~~≥ [4500] kW and ≤ [5000] kW~~ ≥ 2340 kW and ≤ 2600 kW (Unit 1) ≥ 3814 kW and ≤ 4238 kW (Unit 2).	

SR that states "Only applicable to Unit 1." ITS SR 3.8.1.5.2 states "Check for and remove accumulated water from each day tank." A Note modifies the SR that states "Only applicable to Unit 2." These changes to the SRs are acceptable because only the Unit 1 DGs have both day and engine mounted tanks that must be checked for water. For Unit 2, the day tank is the appropriate tank to verify because Unit 2 does not have an engine mounted tank.

11. ISTS SR 3.8.1.7 requires the fast start of each DG to rated voltage and frequency every 184 days. This requirement is not added because the units are licensed in accordance with applicable Safety Guide or Regulatory Guide and does not require the performance of this test and a fast start is performed once every 18 months. The Unit 1 DGs will not field flash on fast start unless an undervoltage signal is present. The Unit 1 DGs can not be emergency started from the control room. This change will minimize the fast starts for the DGs. The ISTS SRs that follow are re-numbered to reflect this SR deletion.

12. ISTS SR 3.8.1.11 requires the fast start of each DG to rated voltage and frequency on an actual or simulated loss of offsite power every 18 months. This requirement is not added because the units are licensed in accordance with applicable Safety Guide or Regulatory Guide and do not require the performance of this test and a fast start is performed on a loss of offsite power concurrent with an ESF signal once every 18 months. The loss of offsite power concurrent with an ESF signal performs some of the technical requirements listed in this SR. The ISTS SRs that follow are re-numbered to reflect this SR deletion.

13. ISTS SR 3.8.1.12 requires the fast start of each DG to rated voltage and frequency on an actual or simulated ESF actuation (SI) signal every 18 months. This requirement is not added because the units are licensed in accordance with applicable Safety Guide or Regulatory Guide and do not require the performance of this test and a fast start is performed on a loss of offsite power in conjunction with an ESF signal once every 18 months. This is another fast start of the DG with the machine not loading and the emergency buses continued to be powered from the offsite source. The ISTS SRs that follow are re-numbered to reflect this SR deletion.

14. ISTS SR 3.8.1.14 requires the performance of ≥ 24-hour run for each DG every 18 months. This requires a minimum 2-hour run at 105 % to 110 % of rated load and the remaining time of 90 % to 100 % of rated load. Proposed ITS SR 3.8.1.10 requires ≥ 8-hour DG run for each DG every 18 months. For ≥ the first 2 hours, ITS SR 3.8.1.10 requires the DGs be run at a load of ≥ 2750 kW for Unit 1 and ≥ 4238 kW for Unit 2 up to the 2000 hour load limit for each DG (Unit 1 2850 kW and Unit 2 4535 kW). For the remaining hours of the test, ITS SR 3.8.1.10 requires the DGs be run at a load equivalent to the continuous duty rating of the DG (i.e., ≥ 2340 kW and ≤ 2600 kW for Unit 1 and ≥ 3814 kW and ≤ 4238 kW for Unit 2).

    The proposed change revising the duration of the required DG run time from 24-hours to 8-hours is acceptable because the 8 hour duration is considered sufficient to demonstrate DG operability. This change is based on the requirements of IEEE Standard 387-1995, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations. The specified DG loading for the 2-hour requirement in ITS SR 3.8.1.10 ensures the capability of the DGs to sustain the full emergency loading requirements without excessive loading. The DG

loading specified for the remainder of the 8-hour run is consistent with the continuous duty rating of the DGs for each unit.

15. ISTS SR 3.8.1.15 requires the hot fast re-start of each DG to rated voltage and frequency after operating for at least 2 hours every 18 months. This test is normally associated with the requirement to perform a 24-hour run (to establish the required "hot" conditions). This surveillance requirement is not added to the BVPS ITS because current licensing basis does not require the performance of this test. This is acceptable for Unit 1 because Safety Guide 9, the Unit's current licensing basis, did not require the performance of this surveillance. For Unit 2 this is acceptable because it is the current licensing basis described by the NRC's initial SER and the Unit 2 UFSAR which took exception to such testing requirements as explained in JFD 14 above. The ISTS SRs that follow are re-numbered to reflect this SR deletion.

16. ISTS SR 3.8.1.20 requires the verification that each DG when started simultaneously from a standby condition can achieve rated voltage and frequency within 10 seconds. ITS SR 3.8.1.15 requires the performance of the test for Unit 2 only. A Note is added to SR to specify that it is applicable to Unit 2 only. This is acceptable because the current licensing basis does not require the SR for Unit 1 and not required by Safety Guide 9, the Unit's current licensing basis.

17. Not used.

18. ISTS LCO 3.8.1 states "The following AC electrical sources shall be OPERABLE: Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System, b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and c. Automatic load sequencers for Train A and Train B." ITS LCO 3.8.1 states "The following AC electrical sources and sequencer timer(s)shall be OPERABLE: Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System, b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and c. Automatic load sequencer timer(s) for each required

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	---------- **- NOTES -** 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. 3. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.89. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ----------	
	Verify each DG operates for ≥ 8 hours:	18 months
	a. For ≥ 2 hours loaded ≥ 2750 kW and ≤ 2850 kW (Unit 1) ≥ 4238 kW and ≤ 4535 kW (Unit 2), and	
	b. For the remaining hours of the test loaded ≥ 2340 kW and ≤ 2600 kW (Unit 1) ≥ 3814 kW and ≤ 4238 kW (Unit 2).	

Section 3.5 RAIs

Question 200802121126

Return to View Menu | Print Document

RAI Screening Required: Yes

Status: Closed

This Document will be approved by: **Gerald Waig; Carl Schulten**

Regulatory Basis must be included in Comments section of this Form

This document has been reviewed and information in this question contains NO SUNSI sensitive material (the checkbox to the right must be selected before this question can be submitted)

Yes

NRC ITS TRACKING

NRC Reviewer

ID	**200802121126**		Conference Call Requested? **Yes**	
Category	**In Scope**			
ITS Information	ITS Section: **3.5 Ross Telson**	TB POC:	JFD Number: **1**	Page Number(s): **87**
	ITS Number: **3.5.4**	OSI: **None**	DOC Number: **None**	Bases JFD Number: **3**

Attachment 1, Volume 10, Rev. 0, Pages 87, 88, 92, 94 and 96 of 98,
---Regarding:
ITS SR 3.5.4.1, 3.5.4.3 (pg 87) and ITS BASES B3.5.4 Borated Water Storage Tank (BWST) APPLICABLE SAFETY ANALYSES (pg 92) and SURVEILLANCE REQUIEMENTS (pg 94) vs. JFD-1 (pg 88) B-JFD 3 (pg 96)
---Request:
1. Please clarify JFD-1 (pg 88) with regard to how the analyses and evaluation included in the Davis-Besse safety analysis report is able to support a SR 3.5.4.3 maximum BWST boron concentration of 2800 ppm at a SR 3.5.4.1 minimum water temperature of 35F, 350 ppm higher and 5F lower than the typical B&W plant modeled in the NUREG, which supports a bracketed maximum boron concentration of 2450 ppm at a bracketed minimum temperature of 40F (pg 87).
2. Please clarify B-JFD 3 (pg 96) for substituting the phrase "50% of the" for the words "all" in the phrase "Large break LOCAs assume that all control rods remain withdrawn from the core" (pg 92). Does the analyses and evaluation included in the Davis-Besse safety analysis report for large break LOCA differ this much from the large break LOCA analysis of the typical B&W plant modeled in the NUREG?
3. Given the additional post-LOCA negative reactivity associated the insertion of 50% of the control rods in the large break LOCA analyses and evaluation included in the Davis-Besse safety analysis report, as opposed to all control rods remaining withdrawn in large break LOCA analysis of the typical B&W plant modeled in the NUREG (pg 92), please explain briefly why the proposed ITS SR 3.5.4.3 minimum boron concentration, at 2600 ppm, is 330 ppm higher than the 2270 ppm bracketed minimum in the typical B&W plant modeled in the NUREG. (pg 87)
4. Please resolve the apparent disparity between the retention of NUREG SR 3.5.4.1 BASES (pg 94) and deletion or alteration of similar phrases in the NUREG APPLICABLE SAFETY ANALYSES (pg 92) and confirm which analyses and evaluation included in the safety analysis report form the bases for- and are assured by the proposed ITS SR limits on BWST boron concentration and temperature.
5. Please confirm whether the ITS SR-proposed temperature and boron concentration bands

Comment	**include appropriate margin to accommodate instrument uncertainty, drift, etc. What is the magnitude of those margins? If not accommodated in the TS SR acceptance criteria, please identify the licensee process or program that controls these margins and indicate whether it is subject to 10 CFR 50.59 change control regulation.** **See basis for request for further clarification of above questions.** **---Basis for Request:** **- On Pg. 92, the NUREG APPLICABLE SAFETY ANALYSES phrase "and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis." is stricken from the NUREG phrase "This temperature also helps prevent boron precipitation and assures that water injection in the reactor vessel..." Similarly, the ITS phrase, "is assumed for the containment vessel vacuum breaker sizing," is substituted for the NUREG phrase "was established to ensure that the solution will not freeze." In context, the original NUREG phrase reads: "The 40F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze." In contrast, on Pg. 94, the following NUREG SR 3.5.4.1 BASES phrase remains unaltered: "Verification every 24 hours that the BWST water temperature is within specified temperature band ensures that the boron will not precipitate; the fluid will not freeze; the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis..."** **- The percentage of control rods assumed to insert at Davis-Besse following a large break LOCA differs by 50% from that of the typical B&W plant modeled in the NUREG. Also the B-JFD 3 states only that this change is made to reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.** **- The combination of (1) deviations between the Davis-Besse ITS and the bracketed NUREG SR acceptance criteria values, (2) the associated BASES changes and apparent contradictions within those BASES, and (3) the vague and generic nature of the associated JFD and B-JFD, warrant additional clarification and confirmation that proposed ITS SR acceptance criteria values and BASES are consistent with applicable Davis-Besse CLB analyses and assumptions.** **---Regulatory Requirements:** **§ 50.36 Technical Specifications** **(a) Each applicant for a license ... shall include ... proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications ... shall also be included in the application, but shall not become part of the technical specifications.** **(b) ... The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.** **(c)(3) Surveillance requirements... assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.**
Issue Date	**02/12/2008**
Close Date	**09/02/2008**

Logged in User: Anonymous

Responses

Licensee Response by Bill Bentley on 03/08/2008	Response 1 License Amendment 207 increased the limits for BWST boron to greater than or equal to 2600 ppm and less than or equal to 2800 ppm, as part of increasing the length of a fuel cycle to 24 months. In the NRC Safety Evaluation, dated February 27, 1996, Section 3.4 (ADAMS Accession Number ML021210121), it is stated that "the minimum boron concentration is based on the

	required concentration to ensure a 1% shutdown margin post LOCA. The maximum boron concentration was determined based on a post-LOCA boron precipitation analysis." The staff found the changes to be acceptable. 35 degrees F was the minimum BWST water temperature specified in the original Davis-Besse Technical Specifications. The phrases in question in the Applicable Safety Analyses (ASA) section of the Bases (Volume 10, Page 92) were altered because the only analysis limit related to BWST minimum temperature is for the Inadvertent Containment Spray Actuation Analysis (described near the end of UFSAR Section 6.2.1.3.2). While it appears reasonable and self evident that 35 degrees F will prevent freezing, this specific stipulation is not made in the UFSAR. We have operational guidance to place a heating system in service whenever the BWST reaches 50 degrees F. We maintained the wording concerning precipitation, since boric acid precipitation is a generally understood concern with these types of systems. As an added note, the solubility limit of a boric acid solution at 35 degrees F is approximately 4900 ppm; therefore precipitation is not a concern. As described by the reviewer, similar phrases in the Bases for SR 3.5.4.1 (Page 94) were not changed to match the modified wording in the ASA section of the Bases. Similar changes will also be made as shown in the attached Markup. This change will be reflected in the supplement to this section of the ITS Conversion Amendment. Response 2 The Justification for Deviation (JFD) 3 (Page 96) wording "changes are made which reflect the plant specific licensing basis description" is correct based on the changes made as part of License Amendment 207 (which increased the minimum boron concentration limit to 2600 ppm). When License Amendment 207 was issued, the NRC also issued the Bases as part of a License Amendment. The Bases provided with License Amendment 207 (ADAMS Accession Number ML021210121) for the BWST Specification 3/4.5.4 stated in part "the limits on the BWST minimum boron ensure that the reactor will remain at least 1% subcritial in the cold condition at 70F, xenon free, while only crediting 50% of the control rod's worth following mixing of the BWST and the RCS water volumes." Response 3 The values reflect what was approved by License Amendment 207. As part of changing from an 18 month to a 24 month fuel cycle, the minimum value was changed from 2100ppm to 2600ppm. Since the NUREG is based on a Frequency of 18 months, perhaps this is the reason for the difference between the Davis-Besse value and the bracketed NUREG value. Also, the fact that the value is bracketed in the NUREG means that the value is plant-specific; it is not a common value for all plants. Response 4 As described in the response to NRC Question 1 above, Davis-Besse will change the words in SR 3.5.4.1 to match the wording in the ASA section of the Bases. Response 5 BWST Boron margins were described in a previous question thread (NRC Question 200801021633, Jerry Jones Response on 02/11/08 for Action 2b). BWST Temperature was addressed in the same question for Action 2d. Any changes to the surveillance procedure and calculations are subject to 10 CFR 50.59.
NRC Response by Ross Telson on 05/05/2008	The STS Bases (prepared in accordance with 50.36(a) and (b)) provide "a summary statement of the bases or reasons for" the BWST low temperature specification. They state, in part, that the BWST lower temperature limit ensures "the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis;" a. Please verify that the minimum temperature assumed in the Davis Besse vessel stress analyses is lower than 35 deg-F, and provide the reference. b. Please explain why the notation of the STS Bases need not be included in the Davis Besse TS Bases, as per 10 CFR 50.36 (a) and (b).
Licensee Response by Jerry Jones on 05/07/2008	Davis-Besse has performed additional review of this issue and concluded the following: Items are included in the Technical Specifications (as Limiting Conditions for Operation) as required by 10 CFR 50.36. As described in 10 CFR

	50.36, the Bases for the LCOs and the associated limits should explain how those items support their reasons for being included in the Technical Specifications. Thus, for the BWST, The ISTS Bases provides the following explanations: 1) The ISTS Bases Background Section (Volume 10, Page 90) states that the purpose of the BWST LCO is to ensure that: "a. The BWST contains sufficient borated water to support the ECCS during the injection phase, b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and containment spray pumps at the time transfer to the recirculation mode of cooling, and c. The reactor remains subcritical following a LOCA." 2) The ISTS Bases Applicable Safety Analyses Section, first paragraph (Page 91) states that " During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and containment spray pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section Specifications B 3.5.2, "ECCS - Operating," and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the BWST in order to evaluate their effects in relation to the acceptance limits." 3) The ISTS Bases LCO Section (Page 93) states "The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the containment in the event of a DBA; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains subcritical following a DBA; and to ensure an adequate level exists in the containment sump to support ECCS and containment spray pump operation in the recirculation mode." Nothing in these three Bases descriptions for the BWST state that the Technical Specification purpose for the BWST is to ensure the assumptions of the reactor vessel stress analysis are met. The only place this is located is in the specific description of the limit in the Applicable Safety Analyses Section (Page 92). Davis-Besse notes that all the other descriptions for the specific BWST limits in the Applicable Safety Analyses Section support the reasons provided for including the Specification as quoted above. The Current Technical Specification Bases make no mention of the reactor vessel stress analysis; it only lists reasons similar to those found in the ISTS Bases. Davis-Besse reviewed the other two PWR ISTS Bases for the similar component. Neither of these two NUREG Bases state that the purpose of the limit is to protect the assumptions in the post-LOCA reactor vessel stress analysis. NUREG-1432, CEOG ISTS, Page B 3.5.4-3 states that the lower limit on Refueling Water Tank temperature is the limit assumed in the accident analysis. NUREG-1431, WOG ISTS, Page B 3.5.4-3 states that in the ECCS analysis, the containment spray temperature assumed to be equal to the Refueling Water Storage Tank lower limit. The UFSAR Chapters 6 and 15 analyses are those that are considered with respect to Criterion 2 and 3 of 10 CFR 50.36. The post-LOCA reactor vessel stress analysis is not included within this population. The reactor vessel stress analyses that were performed (including emergency and faulted conditions) are described in UFSAR Chapter 5. Therefore, the Davis-Besse ITS Bases for the lower temperature limit in the Technical Specifications does not need to include any discussion concerning the reactor vessel stress analysis. Davis-Besse has provided the correct reason the temperature is specified in Technical Specification, thus it is appropriate to not include the reactor vessel stress analysis discussion in the Davis-Besse ITS Bases.
NRC Response by Ross Telson on 06/03/2008	A formal RAI letter is currently being processed for expedited release. It will request a formal docketed response. The contents of the draft RAI input are provided below in support of an expedited licensee response. LETTER: By application dated August 3, 2007, FirstEnergy Nuclear Operating Company

proposed to revise the Davis-Besse Unit 1 (D-B) Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) consistent with Improved Standard Technical Specifications (STS) as described in NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants." The Technical Specifications Branch (ITSB) has reviewed the documentation provided by the licensee in connection with the D-B CTS and the proposed ITS deviations from the STS 3.5.4 "BORATED WATER STORAGE TANK (BWST)" BASES. The proposed ITS was evaluated using the guidance in Regulation 10 CFR 50.36, "Technical Specifications," 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," 10 CFR 50.55a, "Codes and Standards," and Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants." Finally, ITSB consulted with the following technical branches having functional authority over various technical and regulatory aspects of this portion of the license amendment request: (1) Vessels and Internals Integrity Branch, (2) Reactor Systems Branch, and (3) Mechanical and Civil Engineering Branch. The ITSB staff, with the concurrence of the above branches, requests additional information (RAI) to complete its review and to verify licensee compliance with the current licensing basis. The ITSB RAI is attached. ATTACHMENT: Request for Additional Information Technical Specifications Branch, NRR Davis-Besse Nuclear Power Station ITS Conversion License Amendment Request Borated Water Storage Tank Minimum Temperature CTS SR 4.5.4.b / ITS SR 3.5.4.1 & Associated BASES Requested Additional Information 1. The following reference is made in the Davis-Besse (D-B) Updated Final Safety Analysis Report (UFSAR) on page 5.2-2, under the heading 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB): "Reactor vessels with lower power ratings but similar geometries and service conditions have been analyzed to demonstrate that the reactor vessel can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. The evaluation is summarized as follows: The state of stress in the vessel during the LOCA was evaluated for an initial vessel temperature of 608 F. The inside of the vessel wall is rapidly subjected to 90 F injection water of the maximum flow rate obtainable. The results show that the integrity of the vessel is not violated." The reactor vessel stress analysis discussed in the above UFSAR statement appears to be inconsistent with the 35 F minimum temperature for the D-B Borated Water Storage Tank (BWST) (a) in the facility's Current Technical Specifications (CTS) and (b) which is being proposed for the Improved Technical Specification (ITS) conversion. a. Provide additional information to clarify the analysis which is being referred to in the above UFSAR quote. b. Does this refer to a stress analysis performed to demonstrate compliance with ASME Code stress limits or was this analysis performed for some other purpose? c. Has the analysis referred to in the UFSAR been performed with the assumption of 35 F injection water? Why or why not? 2. The basis provided for the selection of 35 F as the BWST minimum temperature, in the proposed D-B ITS Bases, is: "The 35 F lower limit on the temperature of the solution in the BWST is assumed for the containment vessel vacuum breaker sizing. This temperature also helps prevent boron precipitation." However, the basis provided for selection of [40 F] as the BWST minimum temperature, in the B&W Design STS Bases, is (emphasis added): "The 40 F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis." Conceptually, the statement from the B&W Design STS Bases appears to be referring to an analysis very similar to that discussed on page 5.2-2 of the D-B UFSAR. If so, a. Explain why

	the analysis referred to in the D-B UFSAR does not need to be re-performed using a 35 F injection water assumption to verify that the basis you suggested for the D-B BWST minimum temperature is the bounding consideration for establishing that limit. Additional Background Information and Regulatory Bases for the RAI The license amendment request (LAR) proposes to revise the D-B CTS to the ITS consistent with STS as described in NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants" as updated by Revision 3.1 to the STS. Specifically, the LAR seeks to adopt STS Bases associated with STS SR 3.5.4.1 with a number of deviations. Among these deviations proposed in the ITS BASES is the omission of the phrase "ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis." 10 CFR 50.36 requires, in part, that TS be derived from the analyses and evaluation included in the safety analysis report and that they be accompanied by a summary statement of the bases or reasons for them. The phrase which the LAR seeks to omit was incorporated in the STS Bases in accordance with 10 CFR 50.36 as one of several summary reasons or bases for the minimum allowed BWST temperature limit. 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," 10 CFR 50.55a, "Codes and Standards," and Appendix A to Part 50, and "General Design Criteria for Nuclear Power Plants," are incorporated into the STS Bases and into the Davis-Besse Current Licensing Basis. Paragraph IV, "The Commission Policy," of 58 FR 39132 "Final Policy on § 50.36 Technical Specifications," clarifies the Commission's expectations regarding the content of the TS Bases. It states, in part: "Each LCO, Action, and Surveillance Requirement should have supporting Bases. The Bases should at a minimum address the following questions and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases… What are the Bases for each Surveillance Requirement and Surveillance Frequency; i.e., what specific functional requirement is the surveillance designed to verify? Why is this surveillance necessary at the specified frequency to assure that the system or component function is maintained, that facility operation will be within the Safety Limits, and that the LCO will be met?"
Licensee Response by Bill Bentley on 07/23/2008	The NRC provided Davis-Besse with a formal written RAI, dated 6/18/2008 (text of the RAI provided in the 6/3/2008 NRC post). In that RAI, the NRC requested additional information concerning the BWST minimum temperature limit in relation to RCS stress analysis. The RAI questions with the anticipated FENOC responses are attached.
Licensee Response by Bill Bentley on 08/15/2008	Original entry text deleted.
Licensee Response by Bill Bentley on 08/15/2008	The NRC provided Davis-Besse with a formal written RAI, dated 6/18/2008 (text of the RAI provided in the 6/3/2008 NRC post). In that RAI, the NRC requested additional information concerning the BWST minimum temperature limit in relation to RCS stress analysis. Based on verbal discussions with the NRC, initial proposed responses to the RAI questions needed further refinement. The RAI questions with the anticipated FENOC responses are attached, and supersede the proposed responses provided on 7/23/2008.
NRC Response by Aron Lewin on 09/02/2008	No further questions at this time. (posted by A. Lewin)

Date Created: 02/12/2008 11:26 AM by Ross Telson
Last Modified: 09/02/2008 08:02 AM

To complete their review, the NRC staff has requested additional information regarding the license amendment application for the conversion of CTS to ITS. The staff request is provided below in bold type followed by the FENOC response for DBNPS.

DBNPS Borated Water Storage Tank (BWST)

1. The following reference is made in the DBNPS Updated Final Safety Analysis Report (UFSAR) on page 5.2-2, under the heading 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB):

"Reactor vessels with lower power ratings but similar geometries and service conditions have been analyzed to demonstrate that the reactor vessel can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. The evaluation is summarized as follows: The state of stress in the vessel during the LOCA was evaluated for an initial vessel temperature of 608 F. The inside of the vessel wall is rapidly subjected to 90 F injection water of the maximum flow rate obtainable. The results show that the integrity of the vessel is not violated."

The reactor vessel stress analysis discussed in the above UFSAR statement appears to be inconsistent with the 35°F minimum temperature for the DBNPS BWST (a) in the facility's CTS and (b) which is being proposed for the ITS conversion.

a. Provide additional information to clarify the analysis which is being referred to in the above UFSAR quote.

The cited UFSAR text is addressing example stress analyses of other reactor pressure vessels (RPVs) that demonstrated continued structural integrity during a Loss of Coolant Accident (LOCA). The differential temperature cited (ΔT= 608-90 = 518°F) is comparable to the differential temperature at DBNPS, where the reactor vessel inlet temperature is 556°F (ΔT = 556-40 = 516°F). The UFSAR is implying that comparable results were obtained in the DBNPS analysis. The paragraphs preceding the cited text describe the methods used by Babcock & Wilcox (now AREVA) to analyze reactor vessels such as Davis-Besse. The assumptions made in the DBNPS analyses are provided by the Davis-Besse Reactor Coolant System (RCS) Functional Specification. For the LOCA transient, that document initially listed an assumed Borated Water Storage Tank (BWST) temperature of 40°F. That was later revised to 35°F, to be consistent with the minimum allowed by current Technical Specification 3.5.4. This is

also the proposed value to be permitted by the Improved Technical Specifications when issued for DBNPS.

b. Does this refer to a stress analysis performed to demonstrate compliance with American Society of Mechanical Engineers Code stress limits or was this analysis performed for some other purpose?

As discussed in the response above, the cited text is describing analyses performed for other RPVs while the paragraphs preceding it describe the DBNPS specific analyses. The DBNPS analyses, including the transient details provided in the RCS Functional Specification, were performed for demonstrating compliance with American Society of Mechanical Engineers (ASME) Code stress limits.

The RCS Functional Specification was revised to include a BWST temperature of 35°F, however the documentation of the basis of the change was not rigorously established. A qualitative extrapolation of conservative technical analyses performed in the early to mid 1980's to assess Pressurized Thermal Shock (PTS) transients was identified that assessed the effect of 35°F water on RCS components.

c. Has the analysis referenced in the UFSAR been performed with the assumption of 35°F injection water? Why or why not?

As discovered while investigating the analyses needed to respond to this Request for Additional Information (RAI), no explicit detailed analysis of the reactor vessel's structural integrity using 35°F has been completed; however, a qualitative extrapolation based on detailed calculations was done. While the qualitative evaluation used to revise the RCS Functional Specification concluded that the vessel's integrity would not be challenged during a PTS scenario when using 35°F BWST water, it also concluded that the estimated vessel material properties would only support this conclusion up to 25 Effective Full Power Years (EFPY) of operation. The document also stated that detailed analyses would not show as large of an effect on plant life.

This newly identified reactor vessel life limit is more restrictive than the Davis-Besse Operating License. This result is inconsistent with the conclusion drawn relative to Davis-Besse's compliance with 10 CFR 50.61 (see next paragraph, below) and is, *therefore, very unlikely to be a realistic representation of the DBNPS RPV.* This inconsistency in plant life has been identified for resolution in the station's Corrective Action Program. It is expected that FENOC's resolution of the associated condition report will reconfirm that the DBNPS RPV will meet all regulatory requirements related to maintaining structural integrity throughout its licensed life.

This expectation is based on an understanding of the DBNPS vessel construction and materials developed in the period since the qualitative evaluation was performed. The DBNPS vessel is built with only circumferential welds, which is an inherently stronger

configuration for a pressure vessel. In evaluating the DBNPS vessel for compliance with 10 CFR 50.61, the latest End Of Life RT_{PTS} value was determined to be 191.3°F. The regulation's screening criterion is for the EOL RT_{PTS} to be less than 300°F. The DBNPS value has significant margin from the screening criterion. This is an indication that the qualitative evaluation based on the original B&W/Areva analysis is extremely conservative and that no actual concern with the DBNPS reactor vessel integrity exists. The corrective actions taken through the Corrective Action Program will resolve the documentation issues.

2. The basis provided for the selection of 35°F as the BWST minimum temperature, in the proposed DBNPS ITS Bases, is:

"The 35°F lower limit on the temperature of the solution in the BWST is assumed for the containment vessel vacuum breaker sizing. This temperature also helps prevent boron precipitation."

However, the basis provided for selection of [40°F] as the BWST minimum temperature, in the B&W Design STS Bases, is:

"The 40°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis."

Conceptually, the statement from the B&W Design STS Bases appears to be referring to an analysis very similar to that discussed on page 5.2-2 of the DBNPS UFSAR. If so,

a. Explain why the analysis referenced in the DBNPS UFSAR does not need to be re-performed using a 35°F injection water assumption to verify that the basis you suggested for the DBNPS BWST minimum temperature is the bounding consideration for establishing that limit.

FENOC agrees that additional analysis is merited to establish that no reactor vessel integrity issues exist at the proposed allowable minimum BWST temperature. Should the value used in the vessel integrity evaluation be 35°F, then that will also form part of the basis for the specified limit. If a lower temperature is used in evaluating the reactor vessel integrity, the proposed ITS basis described above will be the limiting basis for the specification. Any need to alter the ITS Bases as a result of the resolution of the Condition Report will be identified by implementation of the configuration management program.

DBNPS Emergency Diesel Generator (EDG)

1. Provide the loading profile for the Appendix R scenario to demonstrate that the proposed EDG endurance/margin test ensures that the analyzed functions can be performed. Alternately, confirm by calculation that the EDG Appendix R loading after 2 hours will be less than 100 percent of its continuous rating, considering the 60.5 hertz maximum steady state frequency proposed by the licensee in the ITS.

The table below summarizes the steady-state loading on the Emergency Diesel Generator while the EDG is operating at 60.5 Hz during the Appendix R scenario. The first 2 columns of data are as defined in the AC Power System Analysis Calculation, C-EE-015.03-008, Revision 4. That data reflects the most recent updates to the AC Power System Analysis Calculation.

kW Load	Motor kW	Delta kW (60.5 Hz)	Total kW Load (60.5 Hz)
2493 kW	2238 kW	57 kW	2550 kW

From section 4.2.3.1 of C-EE-015.03-008, Revision 4, the following formula is used to calculate the additional EDG load due to the assumed frequency variation:

Delta kW = ((Assumed frequency / 60)^3 - 1) (Motor kW)

When an assumed frequency of 60.5 Hz is substituted in the calculation, the result is as follows:

Delta kW = ((60.5 / 60)^3 - 1) (2238 kW) = 56.418 kW or 57 kW

Therefore, the Total kW Load = 2493 kW + 57 kW = 2550 kW

The value of 2550 kW is less than 100% of the continuous rating (2600 kW).