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Docket Number 50-346

10 CFR 50.90

License Number NPF-3

Serial Number 2960

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United States Nuclear Regulatory Commission
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Subject: Davis-Besse Nuclear Power Station
License Amendment Application to Revise Technical Specifications Regarding Steam and Feedwater Rupture Control System (SFRCS) Instrumentation Setpoints and Surveillance Intervals (License Amendment Request No. 03-0010)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, a license amendment is requested for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS). The proposed changes affect Technical Specification (TS): 3/4.3.2.2, "Instrumentation - Steam and Feedwater Rupture Control System Instrumentation," including Table 3.3-11, "Steam and Feedwater Rupture Control System Instrumentation," Table 3.3-12, "Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints," and Table 4.3-11, "Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements." Related administrative changes are proposed to TS 3/4.3.2.3, "Instrumentation - Anticipatory Reactor Trip System Instrumentation," Table 3.3-17, "Anticipatory Reactor Trip System Instrumentation," and TS 3/4.3.3.1, "Instrumentation - Monitoring Instrumentation - Radiation Monitoring Instrumentation," Table 3.3-6, "Radiation Monitoring Instrumentation."

Approval of the proposed amendment is requested by March 31, 2004. Once approved, the amendment shall be implemented within 120 days.

A001

The proposed changes have been reviewed by the DBNPS Station Review Board and Company Nuclear Review Board.

Should you have any questions or require additional information, please contact Mr. Kevin L. Ostrowski, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

Lew W Myers

MKL

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
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APPLICATION FOR AMENDMENT

TO FACILITY OPERATING LICENSE NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This submittal requests changes to the Davis-Besse Nuclear Power Station Unit Number 1, Facility Operating License Number NPF-3. The statements contained in this submittal, including its associated enclosures and attachments, are true and correct to the best of my knowledge and belief.

I declare under penalty of perjury that I am authorized by the FirstEnergy Nuclear Operating Company to make this request and the foregoing is true and correct.

Executed on: 8/25/03

By: 
Lew W. Myers, Chief Operating Officer

Docket Number 50-346
License Number NPF-3
Serial Number 2960
Enclosure 1

# DAVIS-BESSE NUCLEAR POWER STATION EVALUATION FOR LICENSE AMENDMENT REQUEST NUMBER 03-0010

(47 pages follow)

# DAVIS-BESSE NUCLEAR POWER STATION EVALUATION FOR LICENSE AMENDMENT REQUEST NUMBER 03-0010

**Subject:** License Amendment Application to Revise Technical Specifications Regarding Steam and Feedwater Rupture Control System (SFRCS) Instrumentation Setpoints and Surveillance Intervals

## 1.0 DESCRIPTION

This letter is a request to amend the Davis-Besse Nuclear Power Station, Unit Number 1 (DBNPS) Facility Operating License Number NPF-3.

The proposed changes affect Technical Specification (TS): 3/4.3.2.2, "Instrumentation - Steam and Feedwater Rupture Control System Instrumentation," TS 3/4.3.2.3, "Instrumentation - Anticipatory Reactor Trip System Instrumentation," and TS 3/4.3.3.1, "Instrumentation - Monitoring Instrumentation - Radiation Monitoring Instrumentation."

## 2.0 PROPOSED CHANGE

In summary, the overall purpose of this License Amendment Request is to:

- Revise the Steam and Feedwater Rupture Control System (SFRCS) Instrumentation Technical Specifications (TSs) to clearly identify the appropriate actions to be taken if an SFRCS instrumentation channel's output logic becomes inoperable,
- Relocate the SFRCS Instrumentation Trip Setpoints from the TSs, and
- Decrease the Channel Functional Test frequency from monthly to quarterly for the SFRCS Instrument Channels and make the associated changes to the Trip Setpoint Allowable Values.

The proposed changes affect Technical Specification (TS): 3/4.3.2.2, "Instrumentation - Steam and Feedwater Rupture Control System Instrumentation," including Table 3.3-11, "Steam and Feedwater Rupture Control System Instrumentation," Table 3.3-12, "Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints," and Table 4.3-11, "Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements." Related administrative changes are proposed to TS 3/4.3.2.3, "Instrumentation - Anticipatory Reactor Trip System Instrumentation," Table 3.3-17, "Anticipatory Reactor Trip System Instrumentation," and TS 3/4.3.3.1, "Instrumentation - Monitoring Instrumentation - Radiation Monitoring Instrumentation," Table 3.3-6, "Radiation Monitoring Instrumentation." These changes are described in further detail below.

Associated changes to the TS Bases are being made under the provisions of the DBNPS TS Bases Control program. The affected TS Bases pages are included in Attachment 3 for information.

<u>Table 3.3-11, "Steam and Feedwater Rupture Control System Instrumentation"</u>

Changes are proposed to Table 3.3-11 in order to separate the TS requirements for the SFRCS instrumentation channels and output logic. Separating these TS requirements will provide for the appropriate action to be taken should a failure occur in an SFRCS instrumentation channel or output logic.

It is proposed that a new Functional Unit 5, "Output Logic," be designated in Table 3.3-11, with a "2" in the "Total No. of Channels" column, a "1" in the "Channels to Trip" column, a "2" in the "Minimum Channel Operable" column, and an "18" in the "Action" column. Associated with this new Functional Unit, a new Action Statement, Action 18, is proposed to be added to Table 3.3-11, to read as follows:

> With any component in the Output Logic inoperable, either declare the associated actuated component(s) inoperable, or place the associated actuated component(s) in the SFRCS-actuated position within one hour.

Associated with these changes, the current Functional Unit 5 in Table 3.3-11, "Manual Initiation (Push buttons)," is proposed to be renumbered as Functional Unit 6.

<u>Table 3.3-12, "Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints"</u>

The NRC's NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants," Revision 2, does not require trip setpoints to be listed in the instrumentation TS. Rather, the existing trip setpoint Allowable Values are listed. As a result, the "Trip Setpoint" values for all of the Functional Units are proposed to be relocated from Table 3.3-12 to the DBNPS Updated Safety Analysis Report. The "Trip Setpoint" column heading is proposed to be removed accordingly.

Since the "Trip Setpoint" column is being relocated from Table 3.3-12, the SFRCS Limiting Condition for Operation (LCO) 3.3.2.2, which refers to this column, must be revised. This LCO is proposed to read as follows:

> The Steam and Feedwater Rupture Control System (SFRCS) instrumentation channels shown in Table 3.3-11 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Allowable Values column of Table 3.3-12.

As a result of revised instrumentation trip setpoint calculations and surveillance testing practices regarding Channel Functional Testing and Channel Calibration, the following changes are proposed. The "Allowable Values" specified via Footnote "**" as applicable to Channel Calibration for Functional Unit 1, "Steam Line Pressure - Low," and Functional Unit 3, "Steam Generator Feedwater Differential Pressure - High," are proposed to be removed from the table since the same Allowable Value applicable to Channel Functional Testing is used for Channel Calibration. The Allowable Value specified as applicable to Channel Functional Testing via Footnote "*" for Functional Unit 1 is proposed to be changed from "≥ 591.6 psig" to "≥ 600.2 psig." The Allowable Value specified as applicable to Channel Functional Testing via Footnote "*" for Functional Unit 2 is proposed to be changed from "≥ 16.9 "" to "≥ 17.3 "." The Allowable Value specified as applicable to Channel Functional Testing via Footnote "*" for Functional Unit 3 is proposed to be changed from "≤ 197.6 psid" to "≤ 176.8 psid." Also, the applicability of the Allowable Values for Functional Unit 4, "Reactor Coolant Pumps - Loss of," which are currently specified via Footnote "#" as applicable to Channel Functional Testing and Channel Calibration, are proposed to be specified via Footnote "*" as applicable to Channel

Functional Testing only. With these changes, Footnotes "**" and "#" are no longer used in the table, and can therefore be deleted.

Table 4.3-11, "Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements"

Consistent with the proposed addition of a new Functional Unit 5 to Table 3.3-11, it is proposed that a new Functional Unit 2, "Output Logic," be designated in Table 4.3-11, with an "NA" (Not Applicable) in the "Channel Check" and "Channel Calibration" columns, and an "M" (Monthly) in the "Channel Functional Test" column.

In addition, the Channel Functional Test surveillance intervals for the four Functional Unit 1 Instrument Channels listed in Table 4.3-11 are proposed to be changed from "M" to "Q" (Quarterly). Technical Specification Table 1.2, "Frequency Notation," defines "M" as "At least once per 31 days," and "Q" as "At least once per 92 days."

Associated with these changes, the current Functional Unit 2 in Table 4.3-11, "Manual Actuation," is proposed to be renumbered as Functional Unit 3.

Table 3.3-17, "Anticipatory Reactor Trip System Instrumentation"

Due to the proposed new Table 3.3-11 Action 18, current Actions 18, 19, and 20 in Table 3.3-17 are proposed to be renumbered as Actions 19, 20, and 21, respectively, as an administrative change.

Table 3.3-6, "Radiation Monitoring Instrumentation"

Due to the proposed new Table 3.3-11 Action 18, current Actions 21 and 22 in Table 3.3-6 are proposed to be renumbered as Actions 22 and 23, respectively, as an administrative change.

## 3.0 BACKGROUND

The function of the SFRCS instrumentation TS is to ensure that the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, the specified coincidence logic is maintained, sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and sufficient system functional capability is available for SFRCS purposes from diverse parameters.

The SFRCS is described in the DBNPS Updated Safety Analysis Report (USAR) Section 7.4.1.3. The SFRCS mitigates the release of high energy steam, automatically starts the Auxiliary Feedwater System in the event of a main steam line or main feedwater line rupture, automatically starts the Auxiliary Feedwater System on the loss of both main feed pumps or the loss of all four reactor coolant pumps, and prevents steam generator overfill and subsequent spillover into the main steam lines. The SFRCS also provides a trip signal to the Anticipatory Reactor Trip System (ARTS), and a trip signal to the main turbine.

The SFRCS is comprised of two independent and redundant protection channels. Each channel has its own independent sensors, which are physically separated from the other channel's sensors and from non-safety system components. No communication of any kind exists between either protection channel.

For the purpose of testability and reliability, each SFRCS protection channel is further divided into two sensing channels and two trip logic channels. Each sensing channel provides all monitored digital plant status signals to either logic channel via signal buffer modules, thus making both sets of sensing and logic channels electrically independent. The corresponding output signals of each logic channel are "AND-gated" to form the output logic (actuation) channel.

Signal processing is performed in the SFRCS logic cabinets. The status signal from each corresponding sensing channel monitoring the same plant variable is combined in a "2-out-of-2" logic before it is further processed by the SFRCS logic module. The SFRCS observes continuously two SFRCS sensing and logic channels for a simultaneous trip condition before actual SFRCS trip signals are sent to the SFRCS actuation channel. This makes the actual trip logic of each of the two redundant SFRCS protection channels a "2-out-of-2" logic.

## 4.0 TECHNICAL ANALYSIS

Table 3.3-11, "Steam and Feedwater Rupture Control System Instrumentation"

As previously described, it is proposed that a new Functional Unit 5, "Output Logic," be designated in Table 3.3-11, with a "2" in the "Total No. of Channels" column, a "1" in the "Channels to Trip" column, a "2" in the "Minimum Channel Operable" column, and an "18" in the "Action" column. Associated with this new Functional Unit, a new Action Statement, Action 18, is proposed to be added to Table 3.3-11, requiring that with any component in the output logic inoperable, the associated actuated component(s) be declared inoperable or placed in the SFRCS-actuated position within one hour.

These proposed changes have the desired effect of separating the TS requirements for the SFRCS instrumentation channels and output logic. This approach is consistent with the general approach found in NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants," Revision 2, wherein separate TS requirements are provided for instrumentation channels and output logic, reflecting the different impact each has on instrumentation system operability.

The proposed column entries for the new Functional Unit 5 are consistent with the system design for the SFRCS output logic. The proposed Action 18 is consistent with the approach utilized in NUREG-1430 Specification 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," which, in the event that one or more automatic actuation logic matrices are inoperable, allows one hour to either place the associated component(s) in the engineered safeguard configuration, or to declare the associated component(s) inoperable. Placing the actuated component in its SFRCS-actuated position is equivalent to the output logic performing its safety feature ahead of time. Since the true effect of the output logic inoperability is inoperability of the supported system, an alternative action is to declare the associated component inoperable and enter the required actions of the affected supported system. The proposed one hour time limit for completion of either of these options is consistent with NUREG-1430 Specification 3.3.7. As stated in the NUREG-1430 Specification 3.3.7 Bases, the one hour completion time is based on operating experience and reflects the urgency associated with the inoperabilty of a safety system component. Based on the above, these proposed changes will have no adverse effect on nuclear safety.

The proposed renumbering of the current Functional Unit 5, "Manual Initiation (Push buttons)," to Functional Unit 6 is an administrative change which will have no adverse effect on nuclear safety.

Table 3.3-12, "Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints"

The proposed relocation of the "Trip Setpoint" values for all of the SFRCS Functional Units is consistent with NUREG-1430, which specifies only the Allowable Values for instrumentation Functional Units. Nominal trip setpoints are specified in the setpoint analysis. The SFRCS trip setpoints being relocated from the TS will be incorporated in the DBNPS USAR no later than the implementation date of the requested license amendment. Future changes to these trip setpoints will be performed under the regulatory controls of 10 CFR 50.59, "Changes, Tests, and Experiments." These changes will be submitted to the NRC in accordance with the USAR revision requirements of 10 CFR 50.71(e) and 10 CFR 50.59(d), as applicable. Based on the above, this change will have no adverse effect on nuclear safety.

The proposed relocation of the "Trip Setpoint" column from Table 3.3-12 requires that SFRCS TS LCO 3.3.2.2 be changed to reflect this removal. The proposed change to TS LCO 3.3.2.2 references the "Allowable Values" column of Table 3.3-12 instead of the "Trip Setpoint" column. In addition, since the TS LCO will now only refer to the "Allowable Values" column, the previous exception of the Steam Generator Level - Low Functional Unit referencing the "Allowable Values" column in lieu of the "Trip Setpoint" column can be deleted. These changes are administrative changes related to the proposed Table 3.3-12 changes and have no adverse effect on nuclear safety.

Consistent with updated calculations and current setpoint methodology, the "Allowable Value" for Functional Unit 1 ("Steam Line Pressure - Low"), Functional Unit 2 ("Steam Generator Level - Low"), and Functional Unit 3 ("Steam Generator Feedwater Differential Pressure - High") are proposed to be changed from "≥ 591.6 psig" to "≥ 600.2 psig," from "≥ 16.9 """ to "≥ 17.3 "," and from "≤ 197.6 psid" to "≤ 176.8 psid," respectively.

For the Steam Line Pressure - Low setpoint calculation, the analytical limit remained unchanged at 600 psia (585.6 psig). Adding instrument string uncertainty, including drift, to the analytical limit, a nominal Trip Setpoint of 620 psig was calculated. The Allowable Value was calculated by adding the uncertainty values not tested during normal surveillance testing to the analytical limit. The only non-zero parameter that is not tested is the switch seismic effect. Therefore, the Allowable Value was calculated to be 600.2 psig.

As noted in TS Table 3.3-12, the Allowable Value for Steam Generator Level - Low is specified as "Actual water level above the lower steam generator tubesheet." Due to fluid density effects and the physical location of the instrument tap, "indicated" level will differ from this "actual" level. For the Steam Generator Level - Low setpoint calculation, the analytical limit remained unchanged at 10 inches "actual" (17.66 inches "indicated"). Adding instrument string uncertainty, including drift, to the analytical limit, a nominal Trip Setpoint of 23.1 inches "indicated" was calculated. The Allowable Value was calculated by adding the uncertainty values not tested during normal surveillance testing to the analytical limit. The non-zero parameters that are not tested are the transmitter effects, seismic effects, and process dependent effects. Therefore, the Allowable Value was calculated to be 17.3 inches "actual" (22.75 inches "indicated").

For the Steam Generator Feedwater Differential Pressure - High setpoint calculation, the analytical limit remained unchanged at 200 psid. Subtracting instrument string uncertainty, including drift, from the analytical limit, a nominal Trip Setpoint of 125 psid was calculated. The Allowable Value was calculated by subtracting the uncertainty values not tested during normal surveillance testing from the analytical limit. The only non-zero parameter that is not tested during surveillance testing is the switch seismic effect. Therefore the Allowable Value was calculated to be 176.8 psid.

These proposed Allowable Values are consistent with updated calculations and current setpoint methodology, providing confidence that the analytical limits will not be violated should an SFRCS actuation occur, and will have no adverse effect on nuclear safety.

The applicability of the Allowable Values for Functional Unit 4, "Reactor Coolant Pumps - Loss of," which is currently specified via Footnote "#" as applicable to Channel Functional Testing and Channel Calibration, is proposed to be specified via Footnote "*" as applicable to Channel Functional Testing. There is no distinction between the Channel Functional Test and Channel Calibration for this instrument loop. Thus, one Allowable Value for the entire instrument loop is all that is necessary for this instrument string. The proposed changes reflect current surveillance testing practices and will have no adverse effect on nuclear safety.

The "Allowable Values" specified via Footnote "**" as applicable to Channel Calibration for Functional Unit 1, "Steam Line Pressure - Low," and Functional Unit 3, "Steam Generator Feedwater Differential Pressure - High," are proposed to the removed from the table. For transmitters located in an inaccessible area due to plant conditions in that area (e.g., high temperature or high radiation), the Channel Functional Test would generally not include the transmitter. However, in the cases where the transmitter is accessible, the Channel Functional Test would generally be included in the Channel Calibration, and one Allowable Value for the entire instrument loop is all that is necessary. Since these SFRCS instrument strings are accessible during normal operation, a distinction between Allowable Values for Channel Functional Testing and Channel Calibration, or a combination thereof, is unnecessary. The proposed changes reflect current surveillance testing practices and will have no adverse effect on nuclear safety.

Based on the other proposed changes to Table 3.3-12, Footnotes "**" and "#" are no longer used. The deletion of these footnotes is therefore an administrative change, and will have no adverse effect on nuclear safety.

Table 4.3-11, "Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements"

The proposed addition of a new Functional Unit 2, "Output Logic," to Table 4.3-11 is similar to the proposed change to Table 3.3-11. This change has the desired effect of separating the TS requirements for the SFRCS instrumentation channels and output logic. As previously mentioned, this approach is consistent with the general approach found in NUREG-1430, wherein separate TS requirements are provided for instrumentation channels and output logic, reflecting the different impact each has on instrumentation system operability. Based on the above, these changes will have no adverse effect on nuclear safety.

The proposed renumbering of the current Table 4.3-11 Functional Unit 2, "Manual Actuation," to Functional Unit 3 is an administrative change which will have no adverse effect on nuclear safety.

There are no Channel Check or Channel Calibration requirements applicable to the output logic. Hence the proposed column entries of "NA" (Not Applicable) for the new Functional Unit 2 are appropriate. In addition, the proposed monthly Channel Functional Test surveillance interval requirement for the output logic is not a change to the existing Technical Specification since the output logic testing requirement is presently enveloped by Functional Unit 1, which is currently a monthly requirement. Channel Functional Testing for the output logic will continue to be performed on a monthly surveillance interval, therefore, these changes will have no adverse effect on nuclear safety.

The proposed increase in the Channel Functional Test surveillance interval for the four Functional Unit 1 Instrument Channels from monthly to quarterly is based on the statistical methodology described in EPRI TR-103335-R1, "Guidelines for Instrument Calibration Extension/Reduction - Revision 1: Statistical Analysis of Instrument Calibration Data," with the following clarifications:

1. The calibration data was taken during a functional check and the instrument was adjusted if the as-found (AF) data was outside the tolerance. In most cases, the as-found and as-left (AL) setpoint are the same, indicative of no adjustment being made.

2. The three-month drift ($D_i$) was calculated using the following formula for a time period for which no adjustments or no replacements were made to the instrument:

   $D_i = AF - AL_{i-3}$

   where AF = as-found setpoint for the current functional check
   $AL_{i-3}$ = as-left setpoint for the functional check 3 months previous

3. Since the duration ($d_i$) of the calculated three-month drift ($D_i$) was not always comprised of the same number of days, and the Technical Specification defines the quarterly interval as 92 days, the calculated three-month drift ($Dc_i$) was corrected based on the following equation:

   $Dc_i = D_i(92/d_i)$

4. Due to the large sample size, an outlier analysis was not done.

5. The data was evaluated for normality.

6. The tolerance interval was calculated for a confidence level of 95%, with 95% of the population contained within the tolerance interval (TI), using the following equation:

   $TI_{(95/95)} = xbar \pm ks$

   where $TI_{(95/95)}$ = tolerance interval for 95%/95%
   xbar = Sample mean
   k = Tolerance factor (95/95)
   s = Sample standard deviation

7. If the sample mean is less than 0.01% of the instrument span, the sample mean was considered zero.

8. The following equations were used to determine the mean and standard deviation:

   Mean = xbar = $( \sum Dc_i ) / N$

   where N = sample count
   i = 1 to N

   Sample standard deviation = $[ (1/(N-1)) (( \sum (Dc_i)^2 ) - N (xbar)^2)]^{1/2}$

   where N = sample count
   i = 1 to N

The method used to determine the three-month drift for the instrument channels analyzed the one-month functional check as-found as-left (AFAL) setpoint data for the subject instruments. The drift analysis determined that the functional check for each trip setpoint may be performed at a quarterly frequency.

A review of system maintenance records for the time period encompassing the drift study data collection period (January 1998 through March 2001) was performed. There were several component failures that appeared to be random in nature, however none of these failures would have prevented the system from performing its function. Therefore, based on this good equipment performance history, no hardware or maintenance enhancements are warranted to support the proposed surveillance interval increase.

In summary, the SFRCS instrumentation drift data has been reviewed and it has been confirmed that the drift over the proposed 92-day (quarterly) surveillance test interval will not cause the trip setpoint to be exceeded beyond the allowable value calculated by the setpoint methodology. On-site records, including calculations, supporting data, and setpoint methodology, are available for NRC inspection. Based on the maintenance records review and the use of the described setpoint methodology, the proposed changes will have no adverse effect on nuclear safety.

Table 3.3-17, "Anticipatory Reactor Trip System Instrumentation"

The proposed changes to renumber the current Table 3.3-17 Actions 18, 19, and 20 as Actions 19, 20, and 21, respectively, are a result of the proposed new Table 3.3-11 Action 18. These are administrative changes that will have no adverse effect on nuclear safety.

Table 3.3-6, "Radiation Monitoring Instrumentation"

The proposed changes to renumber the current Table 3.3-6 Actions 21 and 22 as Actions 22 and 23, respectively, are a result of the proposed new Table 3.3-11 Action 18. These are administrative changes that will have no adverse effect on nuclear safety.

Summary

Based on the detailed technical analyses described above, it is concluded that none of the proposed changes will have an adverse effect on nuclear safety.

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 No Significant Hazards Consideration (NSHC)

The proposed changes would revise Technical Specification (TS) Table 3.3-11, "Steam and Feedwater Rupture Control System Instrumentation," and TS Table 4.3-11, "Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements," to identify the Steam and Feedwater Rupture Control System (SFRCS) output logic as a separate Functional Unit. In addition, the proposed changes would revise TS Table 3.3-12, "Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints," to relocate the "Trip Setpoint" values and also modify the "Allowable Values" entries for Functional Unit 1, "Steam Line Pressure - Low," Functional Unit 2, "Steam Generator Level - Low," and Functional Unit 3, "Steam Generator Feedwater Differential Pressure - High," consistent with updated calculations and current setpoint methodology, and revise the applicability of TS Allowable Values for other SFRCS Functional Units in this table. The proposed changes would also revise TS Table 4.3-11 to change the

Channel Functional Test surveillance requirements for the SFRCS instrument channels from monthly to quarterly, consistent with current methodology. The proposed changes would also make related administrative changes to TS Limiting Condition for Operation (LCO) 3.3.2.2, TS Table 3.3-17, "Anticipatory Reactor Trip System Instrumentation," and TS Table 3.3-6, "Radiation Monitoring Instrumentation."

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes do not change any accident initiator, initiating condition, or assumption, and do not involve a significant change to plant design or operation. In addition, the proposed changes do not increase the likelihood of a malfunction of any plant structures, systems, or components, do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path or alter radiological consequences. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The SFRCS instrumentation setpoint analyses will continue to adequately preserve the margin of safety. In addition, there are no new or significant changes to the initial conditions contributing to accident severity or consequences. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

## 5.2 Applicable Regulatory Requirements/Criteria

The updated instrumentation setpoint calculations have been prepared in accordance with Instrument Society of America (ISA) Standard S67.04, "Setpoints for Nuclear Safety-Related Instrumentation," September 1994, and ISA-RP67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," September 1994. ISA S67.04 Part I - 1994 has been endorsed by the Nuclear Regulatory Commission (NRC) through Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," subject to four listed exceptions and clarifications. The four listed exceptions and clarifications, taken verbatim from RG 1.105, and the DBNPS-specific response to each are as follows:

RG 1.105 Regulatory Position C.1

Section 4 of ISA-S67.04-1994 specifies the methods, but not the criterion, for combining uncertainties in determining a trip setpoint and its allowable values. The 95/95 tolerance limit is an acceptable criterion for uncertainties. That is, there is a 95% probability that the constructed limits contain 95% of the population of interest for the surveillance interval selected.

DBNPS Response to Regulatory Position C.1

The 95/95 tolerance limit methodology is not applied directly to calculations at the DBNPS regarding confidence in equipment uncertainties. Much of the instrumentation is of a vintage that the equipment manufacturer specifications do not include uncertainty confidence data.

As an alternative, a sample of historical calibration data was reviewed for this license amendment application to establish an acceptable confidence in uncertainty values for the Steam Generator Feedwater Differential Pressure - High instrument string. The monthly surveillance records for all eight instruments in both actuation channels (four instruments in each of two logic channels) were included in the review sample, which spanned from January 12, 1998 to March 5, 2001 (38 months). For over 98% of the sample (315 of 321), the as-found readings were within the allowable tolerance of the as-left values, such that the instrument strings did not require any adjustment. In only one case (0.3% of the sample) did the as-found value exceed the current Technical Specification Trip Setpoint of less than or equal 197.6 psid. That value was 200 psid and was attributed to an incorrect adjustment. These results demonstrate that even if equipment uncertainty values were higher than assumed in the setpoint calculation, this effect would be at least partially offset by the margin available from the calibration tolerance which is included in the instrument string calculation.

This license amendment application includes a proposed reduction of the Allowable Value from 197.6 psid to 176.8 psid. Also, as described below, the tolerance interval in the setpoint calculation is for a confidence level of 95% with 95% of the population contained within the tolerance interval. This proposed change provides even greater margin with respect to the analytical limit of 200 psid.

Another factor of note is that in order to provide additional margin and to account for field setting tolerances, a setpoint tolerance is established. Margin is gained because the field device is rarely calibrated with the setpoint at the maximum allowed field setting. Any difference between the maximum allowed field setting and the actual field setting results in increased margin from the analytical limit.

The setpoint verification presently performed for channel calibration is accomplished by applying a pressure at the switch and reading a pressure value from the gauge at the switch when it trips. This setpoint verification encompasses the switch and its trip function and, therefore, provides a high confidence in the accuracy of the trip setpoint setting.

In summary, the intended end result of establishing a tolerance limit criterion for uncertainties (such as 95/95) to ensure an accurate instrumentation response, is met at the DBNPS by means of the calculation methods, instrument string calibration, and setpoint verification.

Similar results were achieved with the Steam Generator Level - Low and Steam Line Pressure - Low devices. For the Steam Generator Level – Low instrument strings, there were 328 monthly surveillance records for the signal monitor setpoint. In this population, there were no instances where the signal monitor required recalibration. This results in the devices being within the setting tolerance 100% of the time. For the Steam Line Pressure - Low instrument strings, there were 327 monthly surveillance records, of which 8 were outside the calibration tolerance and required recalibration. This results in the devices being within the setting tolerance 97.6% of the time. For the Loss of Reactor Coolant Pump instrument strings, there were 1288 monthly surveillance records, and no instances where the devices required recalibration. This results in the devices being within the setting tolerance 100% of the time. None of the monthly surveillance records for the Steam Generator Level - Low, the Steam Line Pressure – Low, or the Loss of Reactor Coolant Pump instrument strings had as-found values that were outside of the existing Allowable Values.

RG 1.105 Regulatory Position C.2

Sections 7 and 8 of Part 1 of ISA-S67.04-1994 reference several industry codes and standards. If a referenced standard has been incorporated separately into the NRC's regulations, licensees and applicants must comply with that standard as set forth in the regulation. If the referenced standard has been endorsed in a regulatory guide, the standard constitutes a method acceptable to the NRC staff of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard if appropriately justified, consistent with current regulatory practice.

DBNPS Response to Regulatory Position C.2

Of the standards listed in Section 7 of Part 1 of ISA-S67.04-1994, Standard ANSI/ISA-S51.1, "Process Instrumentation Terminology," is not known to be incorporated separately into the NRC's regulations nor endorsed in a regulatory guide. However, since this standard addresses only terminology, and has negligible impact on the technical content of the submittal and its associated calculation, its use does not require further justification. None of the other standards listed in Section 7 and none of the standards listed in Section 8 of Part 1 of ISA S67.04-1994 are used as part of the basis for this license amendment request.

RG 1.105 Regulatory Position C.3

Section 4.3 of ISA-S67.04-1994 states that the limiting safety system setting (LSSS) may be maintained in technical specifications or appropriate plant procedures. However, 10 CFR 50.36 states that the technical specifications will include items in the categories of safety limits, limiting safety system settings (LSSS), and limiting control settings. Thus, the LSSS may not be maintained in plant procedures. Rather, the LSSS must be specified as a technical specification-defined limit in order to satisfy the requirements of 10 CFR 50.36. The LSSS should be developed in accordance with the setpoint methodology set forth in the standard, with the LSSS listed in the technical specifications.

DBNPS Response to Regulatory Position C.3

In accordance with Section 4.3 of Part 1 of ISA S67.04-1994, the purpose of a LSSS is to assure that protective action is initiated before the process conditions reach the analytical limit. In addition, the LSSS may be the allowable value, the trip setpoint, or both. The limiting safety system settings are developed in accordance with the setpoint methodology and maintained in the DBNPS Technical Specifications as allowable values. (Note: This license amendment request directly affects the Limiting Condition for Operation portion of the Technical Specifications and not the LSSS portion of the Technical Specifications.)

RG 1.105 Regulatory Position C.4

ISA-S67.04-1994 provides a discussion on the purpose and application of an allowable value. The allowable value is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is considered inoperable and corrective action must be taken in accordance with the technical specifications. The allowable value relationship to the setpoint methodology and testing requirements in the technical specifications must be documented.

DBNPS Response to Regulatory Position C.4

The allowable value relationship to the setpoint methodology and testing requirements in the technical specifications is documented in the setpoint calculation. The setpoint calculation is maintained as part of plant records.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 7.0 REFERENCES

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 254.

2. DBNPS Updated Safety Analysis Report through Revision 23.

3. NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Revision 2, April 2001.

4. EPRI TR-103335-R1, "Guidelines for Instrument Calibration Extension/ Reduction – Revision 1, Statistical Analysis of Instrument Calibration Data," Final Report, October 1998.

5. DBNPS Calculations:

   C-ICE-058.01-001, "RCPM Monitor Setpoint for Loss of a Reactor Coolant Pump," Revision 3.

   C-ICE-083.03-001, "SFRCS Low and High Level Setpoints," Revision 16.

   C-ICE-083.03-003, "Setpoint Determination for SFRCS Low Pressure Trip Switches," Revision 7.

   C-ICE-083.03-004, "Setpoint Determination for SFRCS Differential Pressure Trip Switches," Revision 5.

6. NRC Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Systems," Revision 3, December 1999.

7. Instrument Society of America Standards:

   ISA-RP67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," September 1994.

   ISA-S67.04, "Setpoints for Nuclear Safety-Related Instrumentation," September 1994.

## 8.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages
2. Proposed Retyped Technical Specification Pages
3. Technical Specification Bases Pages

**PROPOSED MARK-UP**
**OF**
**TECHNICAL SPECIFICATION PAGES**

(16 pages follow)

INSTRUMENTATION

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.2 The Steam and Feedwater Rupture Control System (SFRCS) instrumentation channels shown in Table 3.3-11 shall be OPERABLE with their trip setpoints set consistent with the values shown in the ~~Trip Setpoint~~Allowable Values column of Table 3.3-12~~, with the exception of the Steam Generator Level-Low Functional Unit which shall be set consistent with the Allowable Value column of Table 3.3-12~~.

APPLICABILITY: MODES 1, 2 and 3.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2829 Date 4/30/03

ACTION:

a. With a SFRCS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-12, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-11, until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with Table 3.3-12.

b. With a SFRCS instrumentation channel inoperable, take the action shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.2.2.1 Each SFRCS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-11.

4.3.2.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.2.3 The STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIME* of each SFRCS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant channels in a specific SFRCS function as shown in the "Total No. of Channels" Column of Table 3.3-11.

* The Main Steam Isolation Valves (MSIVs) response time is to be the time elapsed from the monitored variable exceeding the trip setpoint until the MSIV is fully closed. The Turbine Stop Valves (TSVs) response time is to be the time elapsed from the main steam line low pressure trip condition until the TSV is fully closed.

## TABLE 3.3-11

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|---|
| 1. Main Steam Pressure Low Instrument Channels* | 2 | 1 | 2 | 16# |
| a. PS 3689B Steam Line 1 Channel 1 | | | | |
| b. PS 3689D Steam Line 2 Channel 1 | | | | |
| c. PS 3689F Steam Line 1 Channel 1 | | | | |
| d. PS 3689H Steam Line 2 Channel 1 | | | | |
| e. PS 3687A Steam Line 2 Channel 2 | | | | |
| f. PS 3687C Steam Line 1 Channel 2 | | | | |
| g. PS 3687E Steam Line 2 Channel 2 | | | | |
| h. PS 3687G Steam Line 1 Channel 2 | | | | |

TABLE 3.3-11 (Continued)

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|---|
| 2. Feedwater/Steam Generator Differential Pressure - High Instrument Channels | 2 | 1 | 2 | 16# |
| a. PDS 2685A Feedwater/Steam Generator 2 Channel 2<br>PDS 2685B Feedwater/Steam Generator 2 Channel 2 | | | | |
| b. PDS 2685C Feedwater/Steam Generator 2 Channel 1<br>PDS 2685D Feedwater/Steam Generator 2 Channel 1 | | | | |
| c. PDS 2686A Feedwater/Steam Generator 1 Channel 1<br>PDS 2686B Feedwater/Steam Generator 1 Channel 1 | | | | |
| d. PDS 2686C Feedwater/Steam Generator 1 Channel 2<br>PDS 2686D Feedwater/Steam Generator 1 Channel 2 | | | | |
| 3. Steam Generator Level - Low Instrument Channels | 2 | 1 | 2 | 16# |
| a. LSLL SP9B8 Steam Generator 1 Channel 1<br>LSLL SP9B9 Steam Generator 1 Channel 1 | | | | |
| b. LSLL SP9A6 Steam Generator 2 Channel 1<br>LSLL SP9A7 Steam Generator 2 Channel 1 | | | | |
| c. LSLL SP9A8 Steam Generator 2 Channel 2<br>LSLL SP9A9 Steam Generator 2 Channel 2 | | | | |

TABLE 3.3-11 (Continued)

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|---|
| 3. Steam Generator Level - Low Instrument Channels (continued) | | | | |
| d. LSLL SP9B6 Steam Generator 1 Channel 2<br>LSLL SP9B7 Steam Generator 1 Channel 2 | | | | |
| 4. Loss of RCP Channels | 2 | 1 | 2 | 16# |
| 5. Output Logic | 2 | 1 | 2 | 18 |
| 6~~5~~. Manual Initiation (Push buttons) | | | | |
| a. Initiate AFPT #1 | 1 | 1 | 1 | 17 |
| b. Initiate AFPT #2 | 1 | 1 | 1 | 17 |
| c. Initiate AFPT #1 and Isolate SG #1 | 1 | 1 | 1 | 17 |
| d. Initiate AFPT#2 and Isolate SG #2 | 1 | 1 | 1 | 17 |

TABLE 3.3-11 (Continued)

TABLE NOTATION

* May be bypassed when steam pressure is below 750 psig. Bypass shall be automatically removed when the steam pressure exceeds 800 psig.

\# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable section of the channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With any component in the Output Logic inoperable, either declare the associated actuated component(s) inoperable, or place the associated actuated component(s) in the SFRCS-actuated position within one hour.

ADDITIONAL CHANGES PREVIOUSLY PROPOSED BY LETTER
Serial No. 2829 Date 4/30/03

## TABLE 3.3-12

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNITS | ~~TRIP SETPOINTS~~ | ALLOWABLE VALUES |
|---|---|---|
| 1. Steam Line Pressure - Low | ~~≥ 591.6 psig~~ | ≥ ~~591.6~~ 600.2 psig*<br>~~≥ 586.6 psig**~~ |
| 2. Steam Generator Level - Low[(1)] | ~~N. A.~~ | ≥ ~~16.9~~ 17.3"* |
| 3. Steam Generator Feedwater Differential Pressure - High[(2)] | ~~≤ 197.6 psid~~ | ≤ 176.8 ~~197.6~~ psid*<br>~~≤ 199.6 psid**~~ |
| 4. Reactor Coolant Pumps - Loss of | ~~High ≤ 1384.6 amps~~<br>~~Low ≥ 106.5 amps~~ | High ≤ 1384.6 amps *~~#~~<br>Low ≥ 106.5 amps *~~#~~ |

(1) Actual water level above the lower steam generator tubesheet.

(2) Where differential pressure is steam generator minus feedwater pressure.

* Allowable Value for CHANNEL FUNCTIONAL TEST

~~** Allowable Value for CHANNEL CALIBRATION~~

~~# Allowable Value for CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION~~

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## TABLE 4.3-11

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST |
|---|---|---|---|
| 1. Instrument Channel | | | |
| a. Steam Line Pressure - Low | S | E | Q~~M~~ |
| b. Steam Generator Level - Low | S | R | Q~~M~~ |
| c. Steam Generator - Feedwater Differential Pressure - High | S | E | Q~~M~~ |
| d. Reactor Coolant Pumps - Loss of | S | E | Q~~M~~ |
| 2. Output Logic | NA | NA | M |
| 3~~2~~. Manual Actuation | NA | NA | R |

INSTRUMENTATION

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.3 The Anticipatory Reactor Trip System instrumentation channels of Table 3.3-17 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-17

ACTION: As shown in Table 3.3-17

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SURVEILLANCE REQUIREMENTS

4.3.2.3 The Anticipatory Reactor Trip System shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the modes and at the frequencies shown in Table 4.3-17.

TABLE 3.3-17

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Turbine Trip | 4 | 2[(a)] | 3 | 1[(b)] | 19~~18~~ |
| 2. Trip of Both Main Feed Pump Turbines | 4 | 2 | 3 | 1 | 20~~19~~ |
| 3. Output Logic | 4 | 2 | 3 | 1 | 21~~20~~ |

(a) Trip automtically bypassed below 45 percent of RATED THERMAL POWER

(b) Applicable only above 45 percent of RATED THERMAL POWER

TABLE 3.3-17 (Continued)

ACTION STATEMENTS

ACTION 19~~18~~ - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or reduce reactor power to less than 45 percent of RATED THERMAL POWER within the next 6 hours.

ACTION 20~~19~~ - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 21~~20~~ - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided both of the following conditions are satisfied:

a) The control rod drive trip breaker associated with the inoperable channel is placed in the tripped condition within one hour.

b) The Minimum Channels OPERABLE requirement is met; however, one additional control rod drive trip breaker associated with another channel may be tripped for up to 2 hours for surveillance testing per Specification 4.3.2.3, after reclosing the control rod drive trip breaker opened in a) above.

## TABLE 4.3-17

## ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. Turbine Trip[a] | S | Not Applicable | SA[c] | 1[b] |
| 2. Main Feed Pump Turbine Trip | S | Not Applicable | SA[c] | 1 |
| 3. Output Logic | Not Applicable | Not Applicable | Q[c] | 1 |

(a) Trip automatically bypassed below 45 percent of RATED THERMAL POWER
(b) Applicable only above 45 percent of RATED THERMAL POWER
(c) Perform on a STAGGERED TEST BASIS

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## 3/4.3.3 MONITORING INSTRUMENTATION

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### RADIATION MONITORING INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes at the frequencies shown in Table 4.3-3.

## TABLE 3.3-6

## RADIATION MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | MEASUREMENT RANGE | ACTION |
|---|---|---|---|---|---|
| 1. AREA MONITORS | | | | | |
| a. Fuel Storage Pool Area Emergency Ventilation System Actuation | 1 | ** | $\leq 2 \times$ background | $0.1 - 10^7$ mr/hr | 23~~22~~ |
| 2. PROCESS MONITORS | | | | | |
| a. Containment | | | | | |
| i. Gaseous Activity RCS Leakage Detection | 1* | 1, 2, 3, & 4 | Not Applicable | $10 - 10^6$ cpm | 22~~21~~ |
| ii. Particulate Activity RCS Leakage Detection | 1* | 1, 2, 3, & 4 | Not Applicable | $10 - 10^6$ cpm | 22~~21~~ |

* As required by Specification 3.4.6.1.
**With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 22~~21~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 23~~22~~ - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

## TABLE 4.3-3

## RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| INSTRUMENT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. AREA MONITORS | | | | |
| a. Fuel Storage Pool Area Emergency Ventilation System Actuation | S | E | M | ** |
| 2. PROCESS MONITORS | | | | |
| a. Containment | | | | |
| i. Gaseous Activity RCS Leakage Detection* | S | E | M | 1, 2, 3 & 4 |
| ii. Particulate Activity RCS Leakage Detection* | S | E | M | 1, 2, 3 & 4 |

* If required by Specification 3.4.6.1 to be OPERABLE.
**With fuel in the storage pool or building

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Attachment 2

**PROPOSED RETYPED**
**TECHNICAL SPECIFICATION PAGES**

(9 pages follow)

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2.2 The Steam and Feedwater Rupture Control System (SFRCS) instrumentation channels shown in Table 3.3-11 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Allowable Values column of Table 3.3-12.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With a SFRCS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-12, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-11, until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with Table 3.3-12.

b. With a SFRCS instrumentation channel inoperable, take the action shown in Table 3.3-11.

### SURVEILLANCE REQUIREMENTS

4.3.2.2.1 Each SFRCS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST during the MODES and at the frequencies shown in Table 4.3-11.

4.3.2.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per REFUELING INTERVAL during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.2.3 The STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM RESPONSE TIME* of each SFRCS function shall be demonstrated to be within the limit at least once per REFUELING INTERVAL. Each test shall include at least one channel per function such that all channels are tested at least once every N times the REFUELING INTERVAL where N is the total number of redundant channels in a specific SFRCS function as shown in the "Total No. of Channels" Column of Table 3.3-11.

* The Main Steam Isolation Valves (MSIVs) response time is to be the time elapsed from the monitored variable exceeding the trip setpoint until the MSIV is fully closed. The Turbine Stop Valves (TSVs) response time is to be the time elapsed from the main steam line low pressure trip condition until the TSV is fully closed.

TABLE 3.3-11 (Continued)

STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|---|---|
| 3. Steam Generator Level - Low Instrument Channels (continued) | | | | |
| d. LSLL SP9B6 Steam Generator 1 Channel 2<br>LSLL SP9B7 Steam Generator 1 Channel 2 | | | | |
| 4. Loss of RCP Channels | 2 | 1 | 2 | 16# |
| 5. Output Logic | 2 | 1 | 2 | 18 |
| 6. Manual Initiation (Push buttons) | | | | |
| a. Initiate AFPT #1 | 1 | 1 | 1 | 17 |
| b. Initiate AFPT #2 | 1 | 1 | 1 | 17 |
| c. Initiate AFPT #1 and Isolate SG #1 | 1 | 1 | 1 | 17 |
| d. Initiate AFPT#2 and Isolate SG #2 | 1 | 1 | 1 | 17 |

TABLE 3.3-11 (Continued)

TABLE NOTATION

* May be bypassed when steam pressure is below 750 psig. Bypass shall be automatically removed when the steam pressure exceeds 800 psig.

# The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable section of the channel is placed in the tripped condition within 1 hour.

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With any component in the Output Logic inoperable, either declare the associated actuated component(s) inoperable, or place the associated actuated component(s) in the SFRCS-actuated position within one hour.

## TABLE 3.3-12

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNITS | ALLOWABLE VALUES |
|---|---|
| 1. Steam Line Pressure - Low | $\geq$ 600.2 psig* |
| 2. Steam Generator Level - Low[(1)] | $\geq$ 17.3"* |
| 3. Steam Generator Feedwater Differential Pressure - High[(2)] | $\leq$ 176.8 psid* |
| 4. Reactor Coolant Pumps - Loss of | High $\leq$ 1384.6 amps *<br>Low $\geq$ 106.5 amps * |

(1) Actual water level above the lower steam generator tubesheet.

(2) Where differential pressure is steam generator minus feedwater pressure.

* Allowable Value for CHANNEL FUNCTIONAL TEST

## TABLE 4.3-11

## STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST |
|---|---|---|---|
| 1. Instrument Channel | | | |
| a. Steam Line Pressure - Low | S | E | Q |
| b. Steam Generator Level - Low | S | R | Q |
| c. Steam Generator - Feedwater Differential Pressure - High | S | E | Q |
| d. Reactor Coolant Pumps - Loss of | S | E | Q |
| 2. Output Logic | NA | NA | M |
| 3. Manual Actuation | NA | NA | R |

TABLE 3.3-17

ANTICIPATORY REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Turbine Trip | 4 | 2[(a)] | 3 | 1[(b)] | 19 |
| 2. Trip of Both Main Feed Pump Turbines | 4 | 2 | 3 | 1 | 20 |
| 3. Output Logic | 4 | 2 | 3 | 1 | 21 |

(a) Trip automtically bypassed below 45 percent of RATED THERMAL POWER

(b) Applicable only above 45 percent of RATED THERMAL POWER

TABLE 3.3-17 (Continued)

ACTION STATEMENTS

ACTION 19 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or reduce reactor power to less than 45 percent of RATED THERMAL POWER within the next 6 hours.

ACTION 20 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided both of the following conditions are satisfied:

a) The control rod drive trip breaker associated with the inoperable channel is placed in the tripped condition within one hour.

b) The Minimum Channels OPERABLE requirement is met; however, one additional control rod drive trip breaker associated with another channel may be tripped for up to 2 hours for surveillance testing per Specification 4.3.2.3, after reclosing the control rod drive trip breaker opened in a) above.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM/TRIP SETPOINT | MEASUREMENT RANGE | ACTION |
|---|---|---|---|---|---|
| 1. AREA MONITORS | | | | | |
| a. Fuel Storage Pool Area Emergency Ventilation System Actuation | 1 | ** | ≤ 2 × background | 0.1 - $10^7$ mr/hr | 23 |
| 2. PROCESS MONITORS | | | | | |
| a. Containment | | | | | |
| i. Gaseous Activity RCS Leakage Detection | 1* | 1, 2, 3, & 4 | Not Applicable | 10 - $10^6$ cpm | 22 |
| ii. Particulate Activity RCS Leakage Detection | 1* | 1, 2, 3, & 4 | Not Applicable | 10 - $10^6$ cpm | 22 |

* As required by Specification 3.4.6.1.
**With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

TABLE NOTATION

| | | |
|---|---|---|
| ACTION 22 | - | With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1. |
| ACTION 23 | - | With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12. |

LAR 03-0010
Attachment 3

**TECHNICAL SPECIFICATION BASES PAGES**

(2 pages follow)

*Note: The Bases pages are provided for information only.*

## BASES

### 3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

The OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensure that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The response time limits for these instrumentation systems are located in the Updated Safety Analysis Report and are used to demonstrate OPERABILITY in accordance with each system's response time surveillance requirements.

For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings b, c, d, e, and f, and Interlock Channel a, and SFRCS ~~Table 3.3-12 Functional Unit 2~~: |

Only the Allowable Value is specified for each Function. Nominal trip setpoints are specified in the setpoint analysis. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip parameter. These uncertainties are defined in the specific setpoint analysis.

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the specified Allowable Values. Any setpoint adjustment shall be consistent with the assumptions of the current specific setpoint analysis.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The frequency is justified by the assumption of an 18 or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.


(Next page is B 3/4 3-1a)

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

The measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The SFRCS RESPONSE TIME for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS RESPONSE TIME ensures that the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

Surveillance Requirement 4.3.2.2.3 requires demonstration that each SFRCS function can be performed within the applicable SFRCS RESPONSE TIME. When this surveillance requirement can not be met due to an inoperable SFRCS-actuated component, the LCO ACTION associated with the inoperable actuated component should be entered. When the SFRCS RESPONSE TIME surveillance requirement can not be met due to inoperable components within the SFRCS, the applicable ACTION ~~16~~ statement of Table 3.3-11 should be followed.

The actuation logic for Functional Units 4.a., 4.b., and 4.c. of Table 3.3-3, Safety Features Actuation System Instrumentation, is designed to provide protection and actuation of a single train of safety features equipment, essential bus or emergency diesel generator. Collectively, Functional Units 4.a., 4.b., and 4.c. function to detect a degraded voltage condition on either of the two 4160 volt essential buses, shed connected loads, disconnect the affected bus(es) from the offsite power source and start the associated emergency diesel generator. In addition, if an SFAS actuation signal is present under these conditions, the sequencer channels for the two SFAS channels which actuate the train of safety features equipment powered by the affected bus will automatically sequence these loads onto the bus to prevent overloading of the emergency diesel generator. Functional Unit 4.a. has a total of four units, one associated with each SFAS channel (i.e., two for each essential bus). Functional Units 4.b. and 4.c. each have a total of four units, (two associated with each essential bus); each unit consisting of two undervoltage relays and an auxiliary relay.

An SFRCS channel consists of 1) the sensing device(s), 2) associated logic and output relays, and 3) power sources. The SFRCS output signals that close the Main Feedwater Block Valves (FW-779 and FW-780) and trip the Anticipatory Reactor Trip System (ARTS) are not required to mitigate any accident and are not credited in any safety analysis. Therefore, LCO 3.3.2.2 does not apply to these functions.

Docket Number 50-346
License Number NPF-3
Serial Number 2960
Enclosure 2

## COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

| COMMITMENTS | DUE DATE |
|---|---|
| List the SFRCS Instrumentation Channel trip setpoints in the USAR. | No later than implementation date of the requested License Amendment. |