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February 20, 2008

L-08-065

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 Application for License Amendment for Measurement Uncertainty Recapture Power Uprate (TAC No. MD5240)

By letter dated April 12, 2007, the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for license amendment. The proposed amendment would revise Technical Specifications for Davis-Besse Nuclear Power Station (DBNPS), Unit No. 1, to accommodate an increase in the Rated Thermal Power from 2772 megawatts thermal (MWt) to 2817 MWt. By letter dated October 3, 2007, the Nuclear Regulatory Commission (NRC) provided a request for additional information concerning FENOC's actions to demonstrate compliance with Title 10 of the Code of Federal Regulations, Part 50 for plant operating conditions when the Leading Edge Flowmeter (LEFM) cannot perform its specified support function in performing the calorimetric heat balance.

By letter dated October 19, 2007, FENOC stated that compliance would be addressed by incorporating a revision to the proposed Technical Specifications. Details of the proposed Technical Specifications are provided in the Enclosure and supplement the initial submittal dated April 12, 2007. An evaluation of the No Significant Hazards Consideration and Environmental Consideration from the initial submittal confirms that these bound the revised Technical Specifications.

A001

NRR

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

Feb 20, 2008.

Sincerely,

Mark B. Bezilla

Enclosure:
Davis-Besse Nuclear Power Station, Unit No. 1, Revised Technical Specifications to Supplement April 12, 2007 Amendment Application

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

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1,
REVISED TECHNICAL SPECIFICATIONS TO SUPPLEMENT
APRIL 12, 2007 AMENDMENT APPLICATION

TABLE OF CONTENTS

1.0 DESCRIPTION

The proposed change would revise the proposed Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), Technical Specifications (TS) originally submitted in correspondence dated April 12, 2007 as follows:

Additional Allowable Values are being added to Table 2.2-1, for the High Flux Functional Unit for operation without the required ultrasonic flow meter instrumentation inputs. Specifically, for operation without the required ultrasonic flow instrumentation and with four pumps operating, the Allowable Value will be $\leq$ 103.3% RTP.

Action 11 is added to Table 3.3-1 for the High Flux Functional Unit to reduce the High Flux trip setpoint within 10 hours after the next required daily heat balance measurement and reduce thermal power when the ultrasonic flow instrumentation is unavailable.

Note 2 of Table 4.3-1 is revised to require use of the ultrasonic flow meter instrumentation when performing the daily secondary heat balance calorimetric unless Action 11 above has been entered.

Attachment 1 contains a proposed mark-up of the affected TS pages while Attachment 2 incorporates the changes in a typed format.

Additionally, Technical Specification Bases changes are included for information only as Attachment 3.

Attachment 4 contains the proposed Technical Requirements Manual page for information only for the required surveillance on the ultrasonic flow meter instrumentation.

2.0 BACKGROUND

As described in correspondence dated April 12, 2007, DBNPS committed to placing administrative controls for the high accuracy instrumentation being used for the Measurement Uncertainty Recovery (MUR) uprate into the Technical Requirements Manual. Resulting from a request for additional information dated October 3, 2007, this change replaces and supplements certain TS pages previously included in License Amendment Application for Measurement Uncertainty Recapture Power Uprate, dated April 12, 2007. The TS change will include requirements to lower power and reset the High Flux trip setpoint if the required high accuracy instrumentation is unavailable. Incorporation of this TS change will reduce the revision to the Technical Requirements Manual as committed to in the initial amendment application.

3.0 TECHNICAL ANALYSIS

FirstEnergy Nuclear Operating Company (FENOC) has evaluated the proposed revision to License Amendment Application for Measurement Uncertainty Recapture Power Uprate, dated April 12, 2007, to determine if the additional changes to the TS will impact the response of the plant or personnel to anticipated transients or accidents. This supplemental response is not introducing or changing any requirements that were considered in the previous submittal. Instead, this supplemented response changes the location of the requirements for instrument unavailability from the licensee-controlled Technical Requirements Manual to the TS. As these changes were evaluated in the previous submittal, the proposed additional changes to the TS are bounded by the analysis submitted in the initial application. Based on the evaluation of the Operating License (OL) and TS changes, it is concluded that the proposed changes will have no adverse effect on nuclear safety.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION

FENOC has evaluated the proposed revision to License Amendment Application for Measurement Uncertainty Recapture Power Uprate, dated April 12, 2007, to determine if the additional changes to the TS will impact the no significant hazards consideration determination submitted in the initial application. The proposed changes to the TS are bounded by the analysis submitted in the initial application. Based on the initial no significant hazards consideration determination bounding the TS revision in this supplement, it is concluded that the proposed license 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 acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

Section 10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

FENOC has reviewed this license amendment application supplement and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment.

6.0 PRECEDENT

License amendment applications based on the use of the Caldon Inc. LEFM CheckPlus™ System were approved for Donald C. Cook Nuclear Plant, Unit 2 (Reference 3) and Seabrook Station, Unit 1 (Reference 4). The aforementioned amendment applications were different from this submittal as the instrument unavailability requirements were located in licensee-controlled documents. Although the requirements may reside in different documents, DBNPS's TS actions to be taken for periods of instrument unavailability meet or exceed the requirements incorporated in the licensee-controlled documents for the aforementioned precedents. Therefore, the referenced precedents are applicable to this submittal.

7.0 REFERENCES

1. FirstEnergy Nuclear Operating Company Letter to NRC, "Davis-Besse Nuclear Power Station License Amendment Application for Measurement Uncertainty Recapture Power Uprate," dated April 12, 2007.

2. NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1 – Request for Additional Information Related to Measurement Uncertainty Recapture Uprate," dated October 3, 2007.

3. NRC Letter to Indiana Michigan Power Company Nuclear Generation Group, "Donald C. Cook Nuclear Plant, Unit 2 – Issuance of Amendment Regarding Measurement Uncertainty Power Uprate (TAC No. MB6751)," dated May 2, 2003.

4. NRC Letter to FPL Energy Seabrook, "Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MC8434)," dated May 22, 2006.

8.0 ATTACHMENTS

1. Proposed Mark-Up of Technical Specification Pages

2. Proposed Retyped Technical Specification Pages

3. Proposed Mark-Up of Technical Specification Bases Pages

4. Proposed Technical Requirements Manual Pages

Docket Number 50-346
License Number NPF-3
Revised TS Supplement
Attachment 1

Revised Proposed Technical Specifications for Measurement Uncertainty Recapture Power Uprate

Contained in this attachment are replacement and supplemental pages for the amendment application previously submitted to the NRC on April 12, 2007. The pages contained in this attachment replaces or supplements specific pages from the initial submittal.

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	~~≤105.1%~~ ≤104.9% of RATED THERMAL POWER with four pumps operating*
	≤80.6% of RATED THERMAL POWER with three pumps operating*
3. RC high temperature	≤618°F*
4. Flux -- Δflux/flow[1]	Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.*
5. RC low pressure[1]	≥1900.0 psig*
6. RC high pressure	≤2355.0 psig*
7. RC pressure-temperature[1]	≥(16.25 T_{out} °F – 7899.0) psig*
8. High flux/number of RC pumps on[1]	≤55.1% of RATED THERMAL POWER with one pump operating in each loop*
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop*
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating*
9. Containment pressure high	≤4 psig*

with secondary heat balance based on ultrasonic flow meter instrumentation.

≤103.3% of RATED THERMAL POWER with four pumps operating with secondary heat balance not based on ultrasonic flow meter instrumentation.

Table 2.2-1. (Cont'd)

(1)Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

This page changed from April 2007 submittal to reflect current approved version of Technical Specifications. Page included for context only.

TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. High Flux	4	2	3	1, 2	2#, 10, 11#
3. RC High Temperature	4	2	3	1, 2	3#, 10
4. Flux - ΔFlux - Flow	4	2(a)(b)	3	1, 2	2#, 10
5. RC Low Pressure	4	2(a)	3	1, 2	3#, 10
6. RC High Pressure	4	2	3	1, 2	3#, 10
7. RC Pressure-Temperature	4	2(a)	3	1, 2	3#, 10
8. High Flux/Number of Reactor Coolant Pumps On	4	2(a)(b)	3	1, 2	3#, 10
9. Containment High Pressure	4	2	3	1, 2	3#, 10
10. Intermediate Range, Neutron Flux and Rate	2	N/A	2(c)	1, 2 and *	4
11. Source Range, Neutron Flux and Rate					
A. Startup	2	N/A	2	2## and *	5
B. Shutdown	2	N/A	1	3, 4 and 5	6
12. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#, 8#
13. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
14. Shutdown Bypass High Pressure	4	2	3	2**, 3** 4**, 5**	6#
15. CR Relays	2	2	2	1, 2 and *	9#

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With one of the Reactor Trip Breaker diverse trip features (undervoltage or shunt trip devices) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

ACTION 9 - With one or both channels of SCR Relays inoperable, restore the channels to OPERABLE status during the next COLD SHUTDOWN exceeding 24 hours.

ACTION 10 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, within one hour, place one inoperable channel in trip and the second inoperable channel in bypass, and restore one of the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and open the reactor trip breakers.

Insert 1

Insert 1

ACTION 11 – In MODE 1 above 50% RATED THERMAL POWER, when the calculated required secondary heat balance is no longer based on ultrasonic flow meter instrumentation,

a. Immediately reduce THERMAL POWER to $\leq$ 98.4% of RATED THERMAL POWER with four reactor coolant pumps operating or to $\leq$ 73.8% of RATED THERMAL POWER with three reactor coolant pumps operating, and

b. Within 10 hours, reduce the High Flux trip setpoint to $\leq$ 103.3% of RATED THERMAL POWER with four reactor coolant pumps operating.

TABLE 4.3-1 (Continued)

Notation

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Insert 2

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP], and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP}\,[API_O - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq 2.5\%$

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

(10) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Safety Analysis Report.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

Insert 2

When > 50% RATED THERMAL POWER, ultrasonic flow meter instrumentation is required to be utilized when performing secondary heat balance calorimetric unless ACTION 11 of Table 3.3-1 is entered.

Docket Number 50-346
License Number NPF-3
Revised TS Supplement
Attachment 2

Proposed Retyped Technical Specifications Pages

Contained in this attachment are replacement and supplemental pages for the amendment application previously submitted to the NRC on April 12, 2007. The pages contained in this attachment replaces or supplements specific pages from the initial submittal.

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	≤104.9% of RATED THERMAL POWER with four pumps operating with secondary heat balance based on ultrasonic flow meter instrumentation*
	≤103.3% of RATED THERMAL POWER with four pumps operating with secondary heat balance not based on ultrasonic flow meter instrumentation*
	≤80.6% of RATED THERMAL POWER with three pumps operating*
3. RC high temperature	≤618°F*
4. Flux -- Δflux/flow[(1)]	Pump allowable values not to exceed the limit lines shown in the CORE OPERATING LIMITS REPORT for four and three pump operation.*
5. RC low pressure[(1)]	≥1900.0 psig*
6. RC high pressure	≤2355.0 psig*
7. RC pressure-temperature[(1)]	≥(16.25 T_{out}°F – 7899.0) psig*
8. High flux/number of RC pumps on[(1)]	≤55.1% of RATED THERMAL POWER with one pump operating in each loop*
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop*
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating*
9. Containment pressure high	≤4 psig*

(1) Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

* Allowable value for CHANNEL FUNCTIONAL TEST.

This page changed from April 2007 submittal to reflect current approved version of Technical Specifications. Page included for context only.

TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. High Flux	4	2	3	1, 2	2#, 10, 11#
3. RC High Temperature	4	2	3	1, 2	3#, 10
4. Flux - ΔFlux - Flow	4	2(a)(b)	3	1, 2	2#, 10
5. RC Low Pressure	4	2(a)	3	1, 2	3#, 10
6. RC High Pressure	4	2	3	1, 2	3#, 10
7. RC Pressure-Temperature	4	2(a)	3	1, 2	3#, 10
8. High Flux/Number of Reactor Coolant Pumps On	4	2(a)(b)	3	1, 2	3#, 10
9. Containment High Pressure	4	2	3	1, 2	3#, 10
10. Intermediate Range, Neutron Flux and Rate	2	N/A	2(c)	1, 2 and *	4
11. Source Range, Neutron Flux and Rate					
A. Startup	2	N/A	2	2 ## and *	5
B. Shutdown	2	N/A	1	3, 4 and 5	6
12. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#, 8#
13. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
14. Shutdown Bypass High Pressure	4	2	3	2**, 3** 4**, 5**	6#
15. CR Relays	2	2	2	1, 2 and *	9#

ACTION STATEMENTS (Continued)

ACTION 8 - With one of the Reactor Trip Breaker diverse trip features (undervoltage or shunt trip devices) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

ACTION 9 - With one or both channels of SCR Relays inoperable, restore the channels to OPERABLE status during the next COLD SHUTDOWN exceeding 24 hours.

ACTION 10 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, within one hour, place one inoperable channel in trip and the second inoperable channel in bypass, and restore one of the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and open the reactor trip breakers.

ACTION 11 - In MODE 1 above 50% RATED THERMAL POWER, when the calculated required secondary heat balance is no longer based on ultrasonic flow meter instrumentation,

a. Immediately reduce THERMAL POWER to $\leq$ 98.4% of RATED THERMAL POWER with four reactor coolant pumps operating or to $\leq$ 73.8% of RATED THERMAL POWER with three reactor coolant pumps operating, and

b. Within 10 hours, reduce the High Flux trip setpoint to $\leq$ 103.3% of RATED THERMAL POWER with four reactor coolant pumps operating.

Notation

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. When > 50% RATED THERMAL POWER, ultrasonic flow meter instrumentation is required to be utilized when performing secondary heat balance calorimetric unless ACTION 11 of Table 3.3-1 is entered.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP], and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP}\left[API_O - API_I\right] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq 2.5\%$

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

(10) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Safety Analysis Report.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

Docket Number 50-346
License Number NPF-3
Revised TS Supplement
Attachment 3

Proposed Mark-Up of Technical Specification Bases Pages

Contained in this attachment are replacement pages for the amendment application previously submitted to the NRC on April 12, 2007. The pages contained in this attachment replace only the specific pages from the initial submittal.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation Allowable Values specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shutdown bypass activated. This high pressure setpoint is lower than the normal low pressure setpoint so that the reactor must be tripped before the bypass is initiated. The high flux setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, except as noted below, a reactor trip is initiated when the reactor power level reaches the Allowable Value ~~≤ 105.1%~~ of 104.9% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be ~~112%~~, 110.2% of rated power which was used in the safety analysis.

Insert 3

Insert 3

The Allowable Values are based on the calculated total loop uncertainty per the methodology documented in the USAR and are the Limiting Safety System Settings as required by 10 CFR 50.36.

The High Flux trip Allowable Value 104.9% RATED THERMAL POWER is based on the assumption that the required high accuracy secondary heat balance instrumentation is necessary to provide sufficient margin between the Reactor Protection System setpoint and Analytical Limits.

With four pumps operating and the Ultrasonic Flow Meter instrumentation not used for heat balance calculations, the High Flux Allowable Value is reduced to 103.3% of RATED THERMAL POWER. This value is calculated by accounting for the difference in heat balance error between the Ultrasonic Flow Meter instrumentation (0.37%) and feedwater venturis (2%).

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

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.

Safety-grade anticipatory reactor trip is initiated by a turbine trip (above 45 percent of RATED THERMAL POWER) or trip of both main feedwater pump turbines. This anticipatory trip will operate in advance of the reactor coolant system high pressure reactor trip to reduce the peak reactor coolant system pressure and thus reduce challenges to the pilot operated relief valve. This anticipatory reactor trip system was installed to satisfy Item II.K.2.10 of NUREG-0737.

Insert 4

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

There are two redundant Fuel Storage Pool Area EVS Area Monitors. With one channel of Fuel Storage Pool Area EVS Area Monitors operable and one channel inoperable, the requirements of TS LCO 3.3.3.1 and TS Table 3.3-6 are satisfied without reliance on the associated actions. Therefore, entry into TS 3.3.3.1 Action b is not required. Appropriate actions with respect to TS 3.9.12 must still be taken.

With zero channels of Fuel Storage Pool Area EVS Area Monitors operable, the requirements of TS LCO 3.3.3.1 and TS Table 3.3-6 are not satisfied, so the TS 3.3.3.1 Action b must be entered. Therefore, Action 22 of Table 3.3-6 must be satisfied.

With one or more of the Containment Activity Monitors (either gaseous or particulate) operable, the requirements of TS LCO 3.3.3.1 and TS Table 3.3-6 are satisfied without reliance on the associated actions. Therefore, entry into TS 3.3.3.1 Action b is not required.

With no Containment Activity Monitors operable, the requirements of TS LCO 3.3.3.1 and TS Table 3.3-6 are not satisfied, so the TS 3.3.3.1 Action b must be entered. Action 21 of Table 3.3-6 must be satisfied.

3/4.3.3.2 INCORE DETECTORS - Deleted

Insert 4

Due to its higher accuracy, the use of Ultrasonic Flow Meter (Leading Edge Flow Meter (LEFM) CheckPlus™ System) instrumentation is preferred for the performance of daily heat balance calculations required by Technical Specification (TS) Surveillance Requirement (SR) 4.3.1.1.1 (Table 4.3-1, Functional Unit 2 - Reactor Protection System High Flux). The use of the LEFM instrumentation for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation provides an uncertainty of 0.37% above 50% of RATED THERMAL POWER (RTP). An uncertainty of 2% is assumed when non-LEFM instrumentation is used for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation. Below 50% of RTP, the heat balance is performed using primary-side instrumentation. Hence, this LCO is only applicable above 50% RTP. In addition, below 73.8% of RTP, the safety analyses have adequate margin to accommodate a 2% heat balance error either with or without the LEFM being used to perform the daily heat balance calculation.

If the LEFM is not available for use, the heat balance will be performed using inputs from less accurate installed instrumentation. Continued power operation is allowed; however, THERMAL POWER must be limited to $\leq$98.4% of RTP with four reactor coolant pumps operating, or $\leq$73.8% of RTP with three reactor coolant pumps operating. Given the larger heat balance uncertainty, these limits preserve the core power used in the USAR accident analysis and the initial conditions for DNB as required by the regulating group operating limits in the COLR.

Also, when operating with four reactor coolant pumps at the reduced power, the Reactor Protection System High Flux trip setpoint Allowable Value must be reduced from $\leq$104.9% to $\leq$103.3% within ten hours of completion of the heat balance calculation using the less accurate instrumentation, in accordance with the requirements of TS 2.2.1. This reduction ensures that when the increased uncertainty of the instrumentation is considered, the maximum analytical setpoint value of 110.2% of RTP will not be exceeded as required by the safety analyses.

Historical comparison of the two feedwater flow measurement systems used for secondary-side heat balance calculations above 50% RTP, LEFM-based and feedwater venturi-based, indicates that the two methods do not diverge significantly during power operations over short periods. The long-term fouling of the venturis results in a more conservative feedwater flow input to the heat balance calculation. Nuclear Instrumentation (NI) trend analysis indicates that the NI to heat balance comparison will not drift significantly over a three-week period, and surveillance data indicates essentially no drift of the high flux setpoints. Accordingly, the accuracy and conservatism of the RPS high flux trip is acceptable in the ten hour period provided for setpoint reduction after completion of the non-LEFM-based heat balance calculation.

The LEFM includes a flow meter measurement section in each of the two main feedwater flow headers. Each measurement section consists of sixteen ultrasonic transducers. With

any transducer inoperable, the Ultrasonic Flow Meter instrumentation system is considered inoperable and the required actions are to be applied.

Docket Number 50-346
License Number NPF-3
Revised TS Supplement
Attachment 4

Proposed Mark-Up of Technical Requirements Manual Pages

Contained in this attachment are replacement pages for the amendment application previously submitted to the NRC on April 12, 2007. The pages contained in this attachment replace only the specific pages from the initial submittal.

3/4.3 INSTRUMENTATION

3/4.3.4 ULTRASONIC FLOW METER INSTRUMENTATION

APPLICABILITY: **MODE 1, when greater than 50% RATED THERMAL POWER**

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ultrasonic Flow Meter instrumentation inoperable.	Restore Ultrasonic Flow Meter to OPERABLE status	Prior to the next required daily calorimetric heat balance measurement.
B. Required Action and associated Completion Time not met.	As shown in Technical Specification 3.3.1.1, Table 3.3-1, Action 11.	Immediately.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
4.3.4.1	The Ultrasonic Flow Meter instrumentation shall be demonstrated OPERABLE by performance of a CHANNEL CHECK.	24 hours

3/4.3.4.1 Ultrasonic Flow Meter Instrumentation

The LEFM includes a flow meter measurement section in each of the two main feedwater flow headers. Each measurement section consists of sixteen ultrasonic transducers. With any transducer inoperable, the Ultrasonic Flow Meter instrumentation system is considered inoperable.

The daily CHANNEL CHECK utilizes the on-line verification and self-diagnostic features of the LEFM to ensure the instrumentation is performing as designed.