APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial Number 2428) concern:

Appendix A, Technical Specifications (TS):

2.2	Limiting Safety System Settings
3/4.3.1	Reactor Protection System Instrumentation
3/4.3.2.2	Steam and Feedwater Rupture Control System Instrumentation
3/4.3.3.5.1	Remote Shutdown Instrumentation
3/4.3.3.6	Post-Accident Monitoring Instrumentation
3/4.4.3	Safety Valves and Pilot Operated Relief Valve - Operating
3/4.4.6.1	Reactor Coolant System Leakage Detection Systems
3/4.7.1.2	Auxiliary Feedwater System

Appendix A, Technical Specification Bases

2.2.1	Reactor Protection System Instrumentation Setpoints
3/4.3.1 and 3/4.3.2	Reactor Protection System and Safety System Instrumentation.

By: John K. Wood
J. K. Wood, Vice President - Nuclear

Sworn to and subscribed before me this 30th day of January, 1997.

Laura A. Jennison
Notary Public, State of Ohio

LAURA A. JENNISON
Notary Public, State of Ohio
My Commission Expires 8-15-2001

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A. Technical Specifications (TS): TS 2.2 Limiting Safety System Settings; TS 3/4.3.1, Reactor Protection System Instrumentation; TS 3/4.3.2.2, Steam and Feedwater Rupture Control System Instrumentation; TS 3/4.3.3.5.1, Remote Shutdown Instrumentation; TS 3/4.3.3.6, Post-Accident Monitoring Instrumentation; TS 3/4.4.3, Safety Valves and Pilot Operated Relief Valve - Operating; TS 3/4.4.6.1, Reactor Coolant System Leakage Detection Systems; TS 3/4.7.1.2, Auxiliary Feedwater System; TS Bases 2.2.1, Reactor Protection System Instrumentation Setpoints; and TS Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation.

A. Time Required to Implement: This change is to be implemented concurrent with related changes to be proposed by separate license amendment applications, prior to the commencement of the Eleventh Refueling Outage (11RFO). The 11RFO is presently scheduled to commence in April, 1998.

B. Reason for Change (License Amendment Request Number 95-0024):

The proposed revisions would modify presently specified 18 month surveillance frequencies in certain TS Surveillance Requirements contained in the above-mentioned TS Sections to new specified frequencies of once each Refueling Interval, or once each 24 months based on the DBNPS Instrument Drift Study. These changes are in accordance with the NRC guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, and will support conversion of the DBNPS from an 18 month to a 24 month fuel cycle.

Setpoint revisions are required by the results of the DBNPS Instrument Drift Study for Steam and Feedwater Rupture Control System instrumentation. These proposed revisions are based on NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995.

Revisions are proposed to Technical Specification 2.2, Limiting Safety System Settings, based on the results of revised Framatome Reactor Protection System instrument string error and allowable value setpoint calculations and are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995.

Several administrative revisions are proposed to instrumentation Technical Specifications and associated TS Bases supporting the preceding areas of revision.

C. Safety Assessment and Significant Hazards Consideration: See Attachment A.

Docket Number 50-346
License Number NPF-3
Serial Number 2428
Attachment A

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 95-0024
(295 pages follow)

**SAFETY ASSESSMENT AND SIGNIFICANT HAZARD CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NO. LAR 95-0024**

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications, to Revise Technical Specifications Regarding Limiting Safety System Settings, Reactor Protection System Instrumentation, Steam and Feedwater Rupture Control System Instrumentation, Remote Shutdown Instrumentation, Post-Accident Monitoring Instrumentation, Safety Valves and Pilot Operated Relief Valve - Operating, Reactor Coolant System Leakage Detection Systems, and Auxiliary Feedwater System for Conversion to a 24 Month Fuel Cycle.

DESCRIPTION:

The Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1 is converting from an 18 month to a 24 month fuel cycle. This conversion will allow the DBNPS to operate at full power for a longer period of time between refueling outages. In order to support this conversion, it is necessary that the DBNPS Operating License NPF-3, Appendix A, Technical Specifications be amended to change the 18 month interval surveillance requirements to 24-month interval surveillance requirements. In addition, the continued applications of TS 4.0.2, which allows surveillance intervals to be increased up to 25% on a non-routine basis, will allow a 24 month surveillance interval to be extended up to 30 months.

License amendment request (LAR) Number 95-0024 addresses only a portion of the scope of changes required for the 24 month cycle conversion. Additional required Technical Specification changes will be submitted under separate but related license amendment applications. Associated changes to the DBNPS Updated Safety Analysis Report (USAR), including the Chapter 15 Accident Analysis, are being evaluated under the 10 CFR 50.59 process. In accordance with 10 CFR 50.59, should this evaluation determine that an unreviewed safety question exists, the USAR changes would be submitted for NRC approval under the license amendment application process.

There are four areas of revision proposed by this LAR:

1. Proposed Revisions to Surveillance Requirement intervals from 18 to 24 months based on the results of the DBNPS Instrument Drift Study.

2. Proposed setpoint revisions required by the results of the DBNPS Instrument Drift Study and based on NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995.

3. Proposed revisions to Technical Specification 2.2, Limiting Safety System Settings, based on the results of revised Framatome Reactor Protection System string error and setpoint allowable value calculations and NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April 1995.

4. Proposed administrative revisions supporting the preceding areas of revision.

Related License Amendment Requests

It is noted that there are other Instrumentation TS Surveillance Requirements, presently on an 18 month frequency under current TS requirements, that are not included in this Safety Assessment and Significant Hazards Consideration and other related LAR's which are associated with the DBNPS conversion to a 24 month fuel cycle. These related LAR's are as follows:

1. License Amendment Request (LAR) 95-0027 (DBNPS letter Serial Number 2405) proposes a revision to TS Definition Table 1.2 redefining Notation "R" from "At least once per 18 months" to "At least once per 24 months" and defining a new Notation "E" as "At least once per 18 months."

   There are several revisions proposed to Surveillance Requirement intervals by this LAR (95-0024) in which the interval is defined by the "R" notation and it is proposed to apply the new definition of the "R" notation discussed in LAR 95-0027.

   All 18 month Surveillance Requirement intervals currently defined by the "R" notation that are to remain at 18 months will be designated by the new "E" notation. All Surveillance Requirement revisions from "R" to "E" notation are proposed under LAR 95-0027 with one exception which is discussed in a subsequent section of this LAR.

2. License Amendment Request 95-0018 (DBNPS letter Serial Number 2342 dated August 7, 1996) proposes a new TS defined term, REFUELING INTERVAL, which is defined as "A period of time $\leq$ 730 days."

   There are several revisions proposed to Surveillance Requirement intervals by this LAR (95-0024) in which the interval is proposed to be defined by this new defined term, REFUELING INTERVAL. Other instrumentation Surveillance Requirement intervals proposed to be revised to REFUELING INTERVAL are proposed in LAR 95-0027.

3. License Amendment Request 96-0014 which addresses the instrument drift study for TS 3.3.2.1 and TS 4.3.2.1.1 (Safety Features Actuation System Instruemtnation), and TS 4.5.2.d (Decay Heat Valve Interlock Channel Functional Test) is scheduled to be submitted to the NRC by May 1997.

Therefore, as noted in the cover letter to this SASHC, Toledo Edison is also requesting that amendments for LAR 95-0018, 95-0027, LAR 96-0014, and this amendment application, LAR 95-0024, be issued together by the NRC.

Proposed Revisions to Surveillance Requirement Intervals

The NRC guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991, (GL 91-04) was utilized in the preparation of this Safety Assessment and Significant Hazards Consideration. Consistent with this guidance, the phrases and similar phrases "at least once per 18 months," and "at least once per 18 months, during shutdown," where applicable, are proposed to be replaced with "At least once each REFUELING INTERVAL." In LAR 95-0018, "REFUELING INTERVAL" is proposed to be defined as "a period of time $\leq$ 730 days" for the 24 month fuel cycle. It is proposed that a new definition for the "R" notation be applied in various TS tables as delineated below. The "R" notation is presently defined by TS Definition Table 1.2 as a frequency of "At least once per 18 months." License Amendment Request 95-0027 proposes that the "R" notation be defined as "At least once per 24 months."

In accordance with the guidance provided by GL 91-04 an instrument drift study was performed for calibration of those instruments that perform safety functions including providing the capability for safe shutdown.

This Safety Assessment and Significant Hazards Consideration (SASHC) proposes a revision to the frequency of several such Surveillance Requirements and addresses the associated instrument drift. These Surveillance Requirements and the specific proposed interval revision (i.e., REFUELING INTERVAL or "R" notation) are individually described in Section 2.A of the enclosures to this SASHC, and include:

1. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 3, RC High Temperature, Channel Calibration.

2. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 4, Flux-ΔFlux-Flow, Channel Calibration as required by table Notation (7).

3. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 5, RC Low Pressure, Channel Calibration.

4. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 6, RC High Pressure, Channel Calibration.

5. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 7, RC Pressure - Temperature, Channel Calibration.

6. 4.3.1.1.1 - Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, Functional Unit 14, Shutdown Bypass High Pressure, Channel Calibration.

7. 4.3.2.2.1 - Table 4.3-11, Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements, Functional Unit 1.b, Steam Generator Level - Low, Channel Calibration.

8. 4.3.3.5.1 - Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements, Instrument 2, Reactor Coolant Temperature-Hot Legs, Channel Calibration.

9. 4.3.3.5.1 - Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements, Instrument 3, Reactor Coolant System Pressure, Channel Calibration.

10. 4.3.3.5.1 - Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements, Instrument 4, Pressurizer Level, Channel Calibration.

11. 4.3.3.5.1 - Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements, Instrument 5, Steam Generator Outlet Steam Pressure, Channel Calibration.

12. 4.3.3.5.1 - Table 4.3-6, Remote Shutdown Monitoring Instrumentation Surveillance Requirements, Instrument 6, Steam Generator Startup Range Level, Channel Calibration.

13. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 1, SG Outlet Steam Pressure, Channel Calibration

14. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 2, RC Loop Outlet Temperature, Channel Calibration.

15. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 3, RC Loop Pressure, Channel Calibration.

16. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 4, Pressurizer Level, Channel Calibration.

17. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 5, SG Startup Range Level, Channel Calibration.

18. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation, Containment High Range Radiation Monitors, Channel Calibration. This surveillance is also affected by a related administrative change discussed in a subsequent section.

19. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 10, RC System Subcooling Margin Monitor, Channel Calibration.

20. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 11, PORV Position Indicator, Channel Calibration.

21. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 13, Pressurizer Safety Valve Position Indicator, Channel Calibration.

22. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 15, Containment Normal Sump Level, Channel Calibration.

23. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 16, Containment Wide Range Water Level, Channel Calibration.

24. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 17, Containment Wide Range Pressure, Channel Calibration.

25. 4.3.3.6 - Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 19, Reactor Coolant Hot Leg Level (Wide Range), Channel Calibration.

26. 4.4.3 - Pilot Operated Relief Valve, Channel Calibration.

27. 4.4.6.1.b - Containment Sump Level and Flow Monitoring System, Channel Calibration.

28. 4.7.1.2.1.d - Auxiliary Feed Pump Turbine Steam Generator Level Control System, Channel Calibration.

29. 4.7.1.2.1.e - Auxiliary Feed Pump Suction Pressure Interlocks, Channel Calibration.

30. 4.7.1.2.2 - Auxiliary Feed Pump Turbine Inlet Steam Pressure Interlocks, Channel Calibration.

Proposed Setpoint Revisions Supporting The Instrument Drift Study

The following revisions are proposed to the Steam and Feedwater Rupture Control System (SFRCS) Allowable Values, Limiting Condition for Operation and Action Statements as a result of the instrument drift study analysis required by GL 91-04:

1. Technical Specification Table 3.3-12, Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints, Functional Unit 2, Steam Generator Level - Low Allowable Values are proposed for revision from "$\geq$ 15.6"*" and "$\geq$ 12.9"**" to "$\geq$ 16.9"*." This new Allowable Value is designated as applicable to the Channel Functional Test only by application of the "*" footnote in Table 3.3-12.

   The Trip Setpoint associated with this Allowable Value is proposed for deletion and is proposed for designation as "N.A."

2. The Limiting Condition for Operation and Action a for TS 3.3.2.2, Steam and Feedwater Rupture Control System Instrumentation, are proposed for revision to reflect the proposed changes to the SFRCS Trip Setpoints and Allowable Values.

   The Limiting Condition for Operation 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 Trip Setpoint column of Table 3.3-12, with the exception of Steam Generator Level - Low Functional Unit which shall be set consistent with the Allowable Value column of Table 3.3-12, and with RESPONSE TIMES as shown in Table 3.3-13."

   Action a. is proposed to read as follows:

   "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."

The proposed revisions to the Allowable Values are a result of the Instrument Drift Study Analysis. Application of the Allowable Values to the Channel Functional Test, deletion of the trip setpoints and proposed revisions to the Limiting Conditions for Operation and Action Statements are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995.

Proposed Revisions to Technical Specification 2.2, Limiting Safety System Settings

The results of the instrument drift study for Reactor Protection System (RPS) instrumentation were incorporated by Framatome into the "Reactor Protection System String Error Calculations" (document 32-1172392-02, dated July 26, 1996) and the "DBNPS Unit 1 RPS Setpoint Allowable Values Calculation" (document 32-1257719-02, dated September 25, 1996). Based on the Framatome calculations the following revisions, which are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995, are proposed:

1. Technical Specification Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints, is proposed for revision to delete the Trip Setpoints, associated column heading and column, and reflect that the existing Allowable Values are applicable to the Channel Functional Test, by application of the "*," footnote in Table 2.2-1 and deletion of the "**," and "#" footnotes to the table.

   For Functional Unit 5, RC low pressure, and Functional Unit 6, RC high Pressure, the Allowable Values applicable to the Channel Calibration, "$\geq$ 1900.0 psig**," and "$\leq$ 2355.0 psig**," respectively, are proposed for deletion.

2. Although not part of the instrument drift study the Allowable Value for TS Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints, Functional Unit 2, RPS High Flux is proposed for revision from "$\leq$ 104.94% of RATED THERMAL POWER with four pumps operating," to "$\leq$ 105.1% of RATED THERMAL POWER with four pumps operating," to reflect the results of the Framatome string error calculations and will also be applicable to the Channel Functional Test, by application of the "*" footnote in Table 2.2-1.

3. Technical Specification 2.2.1, Reactor Protection System Setpoints, and its Action Statement are proposed for revision to reflect the proposed revisions to TS Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints.

   Technical Specification 2.2.1 is proposed to read as follows:

   > "The Reactor Protection System instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1."

   The Action Statement for TS 2.2.1 is proposed to read as follows:

   > "With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to

OPERABLE status with its trip setpoint adjusted consistent with the Allowable Value."

Proposed Administrative Revisions

1. As was identified under "Proposed Revisions to Surveillance Requirement Intervals," Item 18, Technical Specification Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation, is proposed for revision to reflect the proposed revision to 24 month surveillance intervals for the High Range Radiation Monitors and the continued application of an 18 month surveillance frequency for Containment Wide Range Noble Gas Monitors. The proposed revision deletes the existing surveillance interval notations "M" and "R" and identifies two Table entries under Instrument 6, Containment Vessel Post-Accident Radiation as follows:

   Instrument 6.a, Containment High Range Radiation, with a Channel Check Frequency of "M" and a Channel Calibration Frequency of "R." Where "R" reflects the proposed definition in License Amendment Request 95-0027 (DBNPS letter Serial Number 2405) of "At least once per 24 months."

   Instrument 6.b, Containment Wide Range Noble Gas, with a Channel Check Frequency of "M" and a Channel Calibration Frequency of "E." License Amendment Request 95-0027 (DBNPS letter Serial Number 2405) proposes that the "E" notation be defined as "At least once per 18 months."

2. The following revisions are proposed to TS Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, to reflect the changes to TS Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints:

   The first and second paragraph are proposed to read:

   "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 $\leq$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."

The second Paragraph under High Flux is proposed to read as follows:

"During normal station operation, reactor trip is initiated when the reactor power level reaches the Allowable Value ≤105.1% 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%, which was used in the safety analysis."

The first and second paragraphs under Flux - ΔFlux/Flow are proposed to read as follows:

"The power level Allowable Value produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of coolant pumps on trips.

The power level Allowable Value produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate."

The second and third paragraph under RC Pressure-Low, High, and Pressure Temperature are prosed to read as follows:

"During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux setpoint. The Allowable Value for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, ≤2525 psig. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature (16.00 T_{out}-7957.5) psig, Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB."

The first paragraph under Containment High Pressure is proposed to read as follows:

"The Containment High Pressure Allowable Value $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip."

3. Technical Specification Bases 3/4.3.1 and 3/4.3.2 Reactor Protection System and Safety System Instrumentation is proposed for revision to reflect: the proposed revisions to Limiting Safety System Settings and RPS, and SFRCS Allowable Values, Trip Setpoints, Limiting Conditions for Operation and Action Statements; and clarify the requirements for Channel Functional Tests and Channel Calibrations by addition of the following after the third paragraph:

"For the RPS and SFRCS Table 4.3-11 Functional Unit 1.b:

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."

Each of the proposed revisions are shown on the attached marked-up Operating License pages.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

Systems Affected

The Systems Affected by the proposed revisions are identified in item 1.B of Enclosures 1 through 7 to this LAR. Each Enclosure corresponds to a different affected system and its corresponding affected Technical Specifications.

Components Affected

The components affected by the proposed revisions can be discussed in a generic fashion since the instrument uncertainty is applied to the instrument strings in a generic manner with minimal regard to the specific instrument string involved. The instrument strings involved can be categorized into three general types:

1. A process sensing switch with a digital (contact state) output.

2. Process transmitter and indication.

3. A process transmitter, signal conditioning circuitry, bistables, a controller, and/or indication.

A list of affected instrument strings and corresponding Technical Specifications is provided in the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1) on page 14 of 19. A summary of the Instrument Drift Analysis, as performed on each of the affected strings, is contained in Section 3 of Enclosures 1 through 7 to this LAR.

Activities Affected

The basic activity affected by these proposed revisions is the performance of certain surveillance tests on a 24 month frequency instead of an 18 month frequency for each of the systems addressed in Enclosures 1 through 7 of this LAR. The scope of the affect on the surveillance tests, as made necessary by the proposed revisions simply involves either an extension in the interval of performance or an extension of the interval of performance with the application of a modified setpoint, due to a revised Allowable Value. Additionally, there is no actual change in the manner in which the surveillance test is performed due to the proposed revisions. Note that each of the individual Tests affected have been individually identified as a part of the Instrument Drift Study.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

Functions of the Systems Affected

The Functions of the Systems Affected by the proposed revisions are discussed in item 2.B of Enclosures 1 through 7 of this LAR.

Functions of the Components Affected

The Functions of the Components Affected by the revisions proposed in this LAR can be discussed in generic fashion since the application of instrument uncertainty is applied to instrument functions in a generic manner with minimal regard to the specific instrument string involved. The functions of the instrument strings involved can again be categorized according to the same three s listed in the Components Affected section:

1. For the process sensing switch string the switch assembly senses the process (e.g., pressure or level). When the process value goes above or below a prescribed setpoint, the switch provides a digital (i.e., contact state) output to indication, logic circuitry, or equipment actuation.

2. For the transmitter with indication string, the transmitter senses the process (e.g., pressure, temperature, radiation, or differential pressure) and generates a voltage or current signal proportional to the process. This signal is then conditioned (e.g., converted, scaled, or isolated) for further use. The conditioned signal can then be provided as process indication.

3. For the string consisting of transmitter, signal conditioning circuitry, bistable, controller, and/or indication, the transmitter senses the process (e.g., pressure, temperature, radiation, or differential pressure) and generates a voltage or current signal proportional to the process. This signal is then conditioned (e.g., converted, scaled, or isolated) for further use. The conditioned signal can then be applied to a bistable, which when above or below a prescribed setpoint, provides a digital (i.e., contact state) output to various logic circuitry or actuation equipment, or applied to a controller.

In addition, the functions of each of the instrument strings covered by the Instrument Drift Study are discussed in either, the system description found in item 2.B of Enclosures 1 through 7, or the summary of the individual Instrument Drift Analyses in Attachment 3 to this LAR, Enclosures 1 through 7 Section 3.

Functions of the Activities Affected

The Technical Specification Surveillance Tests affected function is that of verifying compliance with specific criteria (such as Allowable Values) as indicated in the Technical Specifications. No changes, are being proposed to the type of testing currently being performed, only to the

length of the surveillance test interval.

INSTRUMENT DRIFT ANALYSIS:

Enclosure 2 of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Support a 24-Month Fuel Cycle," dated April 2, 1991, identifies seven issues to be addressed in justifying increased surveillance intervals to accommodate a 24-month fuel cycle.

Note: Throughout this document the term drift refers to the difference between as-found calibration data and the corresponding as-left data from the previous calibration. The only notable exceptions to this are: item 15b of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS; anywhere drift refers to an uncertainty term in a setpoint or instrument string error calculation; and when used in reference to vendor specifications. For these exceptions, drift is used as defined in ANSI/ISA - S51.1, "Process Instrumentation Terminology."

1. "Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval."

   Calibration data was reviewed for those surveillances proposed for revision to a 24 month interval. As-found and as-left data has not exceeded acceptable limits except on rare occasions. The enclosures to this LAR discuss those occasions where acceptable limits were exceeded.

2. "Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data."

   Drift values were determined for each instrument string. Basic statistics calculated for each drift value included a 95/95% tolerance factor and a 95/95% tolerance interval. Potential data outliers were identified in accordance with ANSI/ASTM E178-1994, "Standard Practice for Dealing With Outlying Observations." Normality of the drift data was verified by the W or D' test in accordance with ANSI N15.15-1974, "Assessment of the Assumption of Normality (Employing Individual Observed Values)." When the test results indicated that the assumption of normality should be rejected a histogram was developed to verify that the drift data was bounded by a normal distribution.

   Time dependency was determined for each instrument string by application of various methods described in Step 13 of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1). If the results of the evaluation were inconclusive then some time dependency was assumed to exist.

3. "Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of channels by TS section that identifies these instrument applications."

   The projected 95/95% tolerance interval for a 30 month interval was determined as described in Step 14 of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1). If the drift was determined to be time-independent then the previously calculated 95/95% tolerance interval was applied to the 30 month interval. If the drift was determined to exhibit time-dependency then the 30 month interval drift was extrapolated for each individual drift data point with a calibration interval less than 30 months.

   The list of affected instrument strings is provided in the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1) page 14 of 19.

4. "Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in a revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that the safety limits and safety analysis assumptions are not exceeded."

   For instrument strings that perform an automatic protective function an analysis of the calculation which established the existing setpoint was performed. Step 15 of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1) describes the methods used to perform these evaluations.

   Setpoint changes are proposed for the Allowable Values for TS 3.3.2.2, Steam and Feedwater Rupture Control System Instrumentation (SFRCS) Table 3.3-12, Functional Unit 2 (Steam Generator Level - Low) based on the results of the associated drift studies.

5. "Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation."

   As discussed in the response to question 4, Step 15 of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1) describes the methods used to perform these evaluations. For instrument strings that provide process variable indication an evaluation has been performed to determine that the instrument could still be utilized to effect a plant shutdown. Enclosures 1 through 7 summarize the evaluations performed to confirm that the instrument errors caused by drift are acceptable to effect safe shutdown of the plant.

6. "Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations."

   The applicable surveillance and periodic test procedures were reviewed to verify that they appropriately reflect all applicable conditions and assumptions of the setpoint and safety analysis. In one case, Auxiliary Feedwater Pump Turbine Inlet Steam Pressure Interlocks, a Potential Condition Adverse to Quality Report was initiated to address deficiencies in procedures.

7. "Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety."

   Appropriate DBNPS surveillance procedures will be revised to require that for any instrument string which had its calibration interval extended from 18 to 24 months a Potential Condition Adverse Quality Report (PCAQR) be initiated if as-found data for any component or string exceeds its expected value limit, as determined consistent with the drift study methodology.

   Following each refueling outage an evaluation will be performed of each PCAQR initiated during that outage to verify that instrument drifts occurring over the increased calibration interval are consistent with the 95/95 percent tolerance intervals established by the drift study and any applicable setpoint and safety analyses. This evaluation will be documented in a revision to the appropriate DBNPS procedures.

The enclosures to this SASHC discuss the results of the instrument drift study.

EFFECTS ON SAFETY:

Proposed Revisions to Surveillance Requirement Intervals

The enclosures to this SASHC describe the effect on safety due to increasing certain surveillance test intervals from 18 to 24 months and the continued application of TS 4.0.2 (which allows surveillance intervals to be increased up to 25% on a non-routine basis). As discussed in Generic Letter 91-04 an instrument drift study analysis was conducted for each affected instrument string. Where projected instrument drift errors exceeded the allowance for instrument drift that was used to establish setpoints either a new setpoint has been established or the calculations excess margin was reduced. Where projected instrument error of instrument strings that provide process variable indication could adversely affect operator control of plant parameters to effect a safe shutdown an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect. This will result in no adverse effect on safety.

In addition, the licensing basis was reviewed for each proposed revision to ensure it was not invalidated. Applicable surveillance data and maintenance history reviews were performed.

Based on the results of the instrument drift study analysis and surveillance data and maintenance history reviews it is concluded that there is no adverse effect on nuclear safety due to increasing the surveillance test intervals from 18 to 24 months and the continued application of TS 4.0.2. In addition, the licensing basis remains valid.

Proposed Setpoint Revisions Supporting The Instrument Drift Study

1. The SFRCS Steam Generator Level - Low instrument strings were rescaled from a 0 to 250 inch range to a 0 to 300 inch range during the Tenth Refueling Outage. This range change was required due to the increased fouling of the Steam Generators (SG). As the SG foul the indicated levels increase. There was no physical change to the components in the affected instrument strings. The results of the drift study were not compared to the existing design basis calculation but were incorporated into the new design basis calculation in accordance with the ISA S67.04 Part I-1994, "Setpoints for Nuclear Safety-Related Instrumentation," and ISA RP67.04, Part II-1994, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," to incorporate the drift study results based on the 0 to 250 inch range calibration data. This resulted in a revision to the existing Allowable Value but did not affect the existing field trip setpoint. The new Allowable Value is conservative with respect to the existing Allowable Values requiring SFRCS actuation at a higher SG level. Therefore, there is no adverse effect on nuclear safety.

   Applicability of the new Allowable Value to the Channel Functional Test only and the proposed deletion of the Trip Setpoint is consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995, and therefore will have no adverse effect on nuclear safety.

2. The proposed revision to the Limiting Condition for Operation and Action Statement "a" for TS 3.3.2.2, SFRCS Instrumentation, associated with the proposed revision to the Allowable Values for SFRCS Steam Generator Level - Low are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995, and therefore will have no adverse effect on nuclear safety.

Proposed Revisions to Technical Specification 2.2, Limiting Safety System Settings

The proposed deletion of the Trip Setpoints, deletion of the Allowable Values applicable to the Channel Calibration for RC low pressure, and RC high pressure functional units, application of all table Allowable Values to the Channel Functional Test as opposed to the Channel Calibration, and

deletion of the "**" and "#" footnotes for Technical Specification Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints, and the proposed revision to TS 2.2, Limiting Safety System Settings, and the associated Action Statement are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995. Therefore, the proposed revisions will have no adverse effect on nuclear safety.

The proposed revision to the Allowable Value for TS Table 2.2-1, Functional Unit 2, High Flux, is based on a revision in how the bistable error is determined and accounted for in the setpoint calculation. This calculation conforms with ISA S67.04, Part I-1994, "Setpoints for Nuclear Safety-Related Instrumentation," and provides a lower bistable error and a slightly higher Allowable Value. The proposed Allowable Value was determined in accordance with the approved setpoint methodology described in Babcock and Wilcox document BAW-10179P, "Safety Criteria for Acceptable Cycle Reload Analyses," and is bounded by the High Flux trip of 112% rated power assumed in the DBNPS accident analysis. Therefore, this proposed revision will have no adverse effect on plant safety.

Proposed Administrative Revisions

1. The proposed revision to Technical Specification Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation separates the radiation monitors to reflect the revision to 24 month surveillance intervals for the High Range Radiation Monitors and that the Containment Wide Range Noble Gas monitors will remain on a 18 month surveillance frequency. This is an administrative change and will have no adverse effect on nuclear safety.

2. The proposed revisions to TS Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, reflect the proposed revisions to TS Table 2.2.1, Reactor Protection System Instrumentation Trip Setpoints. These changes are administrative and will have no adverse effect on nuclear safety.

3. The proposed revisions to TS Bases 3/4.3.1 and 3/4.3.2 Reactor Protection System and Safety System Instrumentation, reflect the proposed revisions to Limiting Safety System Settings and RPS, and SFRCS Allowable Values, Trip Setpoints, Limiting Conditions for Operation and Action Statements and clarifies the requirements for Channel Functional Tests and Channel Calibrations. These changes are administrative and will have no adverse effect on nuclear safety.

Manufacturer or vendor maintenance information for the affected components is considered in the DBNPS Preventive Maintenance (PM) Program. The PM Program is being evaluated as a separate activity in support of the conversion from an 18-month to a 24-month fuel cycle. Changes will be made, as necessary, in the PM Program to facilitate a 24-month fuel cycle.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for the subject Technical Specifications (TS): TS 2.2 Limiting Safety System Settings; TS 3/4.3.1.1, Reactor Protection System Instrumentation; TS 3/4.3.2.2, Steam and Feedwater Rupture Control System Instrumentation; TS 3/4.3.3.5.1, Remote Shutdown Instrumentation; TS 3/4.3.3.6, Post-Accident Monitoring Instrumentation; TS 3/4.4.3, Safety Valves and Pilot Operated Relief Valve - Operating; TS 3/4.4.6.1, Reactor Coolant System Leakage Detection Systems; TS 3/4.7.1.2 and Auxiliary Feedwater System. Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

Results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

The proposed revisions to Allowable Values for Steam and Feedwater Rupture Control System Steam Generator Level - Low are conservative with respect to the current Allowable Values and therefore, do not adversely affect previously analyzed accidents.

The application of the Allowable Value to the Channel Functional Test only, the proposed deletion of the Trip Setpoint, and revision of the Limiting Condition for Operation and Action Statement A for TS 3.3.2.2, SFRCS Instrumentation, associated with the proposed revision of the Allowable Values for SFRCS Steam Generator Level - Low are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995. The proposed revisions will have no adverse effect on any previously analyzed accident.

The proposed revision to the Reactor Protection System High Flux Allowable Value was determined in accordance with the approved setpoint methodology described in Babcock and Wilcox document BAW-10179P, Safety Criteria for Acceptable Cycle Reload Analyses, and is bounded by the High Flux trip of 112% rated power assumed in the DBNPS accident analysis.

The proposed deletion of the Trip Setpoints, deletion of the Allowable Values applicable to the Channel Calibration for RC low pressure, and RC high pressure functional units, application of Allowable Values to the Channel Functional Test as opposed to the Channel Calibration, and deletion of the "**" and "#" footnotes for Technical Specification Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints, and the proposed revision to TS 2.2, Limiting Safety System Settings, are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995. The proposed revisions have no adverse effect on any previously analyzed accident.

The proposed revision to Technical Specification Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation separates the radiation monitors to reflect the revision to 24 month surveillance intervals for the High Range Radiation Monitors and that the Containment Wide Range Noble Gas monitors will remain on a 18 month surveillance frequency is an administrative change and does not affect previously analyzed accidents.

The proposed revision to the Technical Specification Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, and Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, are administrative and do not affect previously analyzed accidents.

Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

These revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by these proposed revisions. Existing system and component redundancy is not being changed by these proposed changes. Existing system and component operation is not being changed by these proposed changes and the assumptions used in evaluating the radiological consequences in the DBNPS Updated Safety Analysis Report are not invalidated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because these proposed revisions do not involve any physical changes to systems or components, nor do they alter the typical manner in which the systems or components are operated.

No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

Results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews.

The proposed revisions to Allowable Values for Steam and Feedwater Rupture Control System Steam Generator Level - Low are conservative with respect to the current Allowable Values and do not alter any testing currently being performed.

The application of the Allowable Value to the Channel Functional Test only, the proposed deletion of the Trip Setpoint, and revision of the Limiting Condition for Operation and Action Statement A for TS 3.3.2.2, SFRCS Instrumentation, associated with the proposed revision to the Allowable Values for SFRCS Steam Generator Level - Low are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995. The proposed revisions do not alter any testing currently being performed.

The proposed deletion of the Trip Setpoints, deletion of the Allowable Values applicable to the Channel Calibration for RC low pressure, and RC high pressure functional units, application of Allowable Values to the Channel Functional Test as opposed to the Channel Calibration, and deletion of the "**" and "#" footnotes for Technical Specification Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints, and the proposed revision to TS 2.2, Limiting Safety System Settings, are consistent with NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995. The proposed revisions do not alter any testing currently being performed.

The proposed revision to the Reactor Protection System High Flux Allowable Value was determined in accordance with the approved setpoint methodology described in Babcock and Wilcox document BAW-10179P, Safety Criteria for Acceptable Cycle Reload Analyses, and is bounded by the High Flux trip of 112% rated power assumed in the DBNPS accident analysis and does not alter any testing currently being performed.

The proposed revision to Technical Specification Table 4.3-10, Post-Accident Monitoring Instrumentation Surveillance Requirements, Instrument 6, Containment Vessel Post-Accident Radiation separates the radiation monitors to reflect the revision to 24 month surveillance intervals for the High Range Radiation Monitors and that the Containment Wide Range Noble Gas monitors will remain on a 18 month surveillance frequency is an administrative change and does not alter any testing currently being performed.

The proposed revision to the Technical Specification Bases 2.2.1, Reactor Protection System Instrumentation Setpoints, and Bases 3/4.3.1 and 3/4.3.2, Reactor Protection System and Safety System Instrumentation, are administrative and do not alter any testing currently being performed.

3. Not involve a significant reduction in a margin of safety because The results of the instrument drift study analysis and review of historical 18 month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or either a new analysis has been performed incorporating a more conservative setpoint or the calculations excess margin was reduced; projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation or an engineering evaluation has been performed to justify continued use of the instrument string and revisions will be made to DBNPS calculations and controlling procedures where appropriate, to offset any adverse effect; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and maintenance records reviews. Existing system and component redundancy is not being changed by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, consequently there are no significant reductions in a margin of safety.

CONCLUSIONS:

On the basis of the above, Toledo Edison has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request involves a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENTS:

Attached are the proposed marked-up changes to the Operating License. Enclosed are summaries of the licensing basis and responses to the Generic Letter 91-04, Enclosure 2 issues for each of the Technical Specifications, Surveillance Requirements and Bases proposed for revision as listed below.

Surveillance data, and maintenance record reviews were performed.

Enclosure 1	Reactor Protection System Instrumentation
Enclosure 2	Steam and Feedwater Rupture Control System Instrumentation
Enclosure 3	Remote Shutdown Instrumentation
Enclosure 4	Post-Accident Monitoring Instrumentation
Enclosure 5	Safety Valves and Pilot Operated Relief - Operating
Enclosure 6	Reactor Coolant System Leakage - Leak detection Systems
Enclosure 7	Auxiliary Feedwater System

Attached is a copy of the "Instrument Drift Data Analysis Methodology and Assumptions" for DBNPS (Attachment 1) which includes a listing of each instrument string and type evaluated in the drift study. Also attached is a copy of the complete instrument drift study for Reactor Protection System, Reactor Coolant Flow (Attachment 2) and a summary of each of the instrument drift study analysis (Attachment 3).

REFERENCES:

1. Davis-Besse Nuclear Power Station (DBNPS) Unit No. 1, Operating License NPF-3, Appendix A, Technical Specifications, through Amendment 212.

2. Davis-Besse Nuclear Power Station Updated Safety Analysis Report through Revision 19.

3. Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991.

4. 10 CFR 50.59, "Changes, Tests, and Experiments."

5. NUREG-1430, Revision 1, "Standard Technical Specifications, Babcock and Wilcox Plants," dated April, 1995.

6. ISA S67.04, Part I-1994, "Setpoints for Nuclear Safety-Related Instrumentation."

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

8. Babcock and Wilcox document BAW-10179P, "Safety Criteria for Acceptable Cycle Reload Analyses."

9. Framatome document 32-1172392-02, "Reactor Protection System String Error Calculations," dated July 26, 1996.

10. Framatome document 32-1257719-02, "DBNPS Unit 1 RPS Setpoint Allowable Values Calculation," dated September 25, 1996.

11. ANSI/ISA - S51.1, "Process Instrumentation Terminology."