Docket Number 50-346
License Number NPF-3
Serial Number 2335
Enclosure 1
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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 2335) concern:

Appendix A, Technical Specification 3/4.3.1.1 - Reactor Protection System Instrumentation, and Technical Specification 3/4.3.2.3 - Anticipatory Reactor Trip System Instrumentation.

By: John P. Stetz
John P. Stetz, Vice President - Nuclear

Sworn to and subscribed before me this 28th day of May, 1996.

Lori J. Strauss
Notary Public, State of Ohio

LORI J. STRAUSS
Notary Public, State of Ohio
My Commission Expires 3/22/98

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The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License Number NPF-3, Appendix A, Technical Specifications. The changes involve Technical Specification 3/4.3.1.1 - Reactor Protection System Instrumentation and Technical Specification 3/4.3.2.3 - Anticipatory Reactor Trip System Instrumentation.

A. Time Required to Implement: This change is to be implemented within 90 days after the NRC issuance of the License Amendment.

B. Reason for Change (License Amendment Request Number 95-0006):

The proposed changes will make the following revisions to Technical Specification 3/4.3.1.1 - Reactor Protection System Instrumentation and Technical Specification 3/4.3.2.3 - Anticipatory Reactor Trip System Instrumentation: Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, for Functional Unit 12, Control Rod Drive Trip Breakers, and Functional Unit 13, Reactor Trip Module Logic, revise the channel functional test frequency from monthly on a staggered test basis to semi-annually on a staggered test basis; Table 4.3-17, Anticipatory Reactor Trip System Instrumentation Surveillance Requirements, for Functional Unit 3, Output Logic, revise the channel functional test frequency from monthly on a staggered test basis to semi-annually on a staggered test basis.

The technical justification for these proposed changes is provided in the Babcock & Wilcox Owners Group Topical Report BAW-10167, Supplement 3, January 1995, "Justification for Increasing The Reactor Trip System On-Line Test Intervals" (BAW-10167).

C. Safety Assessment and Significant Hazards Consideration: See Attachment

Docket Number 50-346
License Number NPF-3
Serial Number 2335
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT
HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NO. 95-0006

(15 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT
HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NO. 95-0006

TITLE:

Proposed Modification to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License NPF-3, Appendix A Technical Specifications to Revise Technical Specification 3/4.3.1.1 - Reactor Protection System Instrumentation and Technical Specification 3/4.3.2.3 - Anticipatory Reactor Trip System Instrumentation to Increase the Trip Device Test Interval

DESCRIPTION:

The purpose of the proposed changes is to increase from monthly to semiannually, on a staggered test basis, the channel functional test surveillance test intervals for the Control Rod Drive Trip Breakers and Reactor Trip Module Logic of the Reactor Protection System Instrumentation, and the Output Logic of the Anticipatory Reactor Trip System Instrumentation. These changes are being proposed because the current Technical Specification required test frequencies, which place the DBNPS in a test configuration susceptible to a spurious reactor trip, have been determined to be unduly restrictive. These changes in test frequencies will put the Reactor Protection System trip module logic on a frequency consistent with the Reactor Protection System instrument string channel functional tests, and put the Anticipatory Reactor Trip System output logic on a frequency consistent with the Anticipatory Trip System turbine trip and main feed pump turbine trip channel functional tests. The technical justification for these proposed changes is provided in B&W Owners Group Topical Report BAW-10167, Supplement 3, January 1995, "Justification for Increasing The Reactor Trip System On-Line Test Intervals."

These changes are also being submitted to the Nuclear Regulatory Commission (NRC) as a Cost Beneficial Licensing Action (CBLA). As demonstrated in BAW-10167, these changes will not adversely impact safety. The DBNPS has experienced three reactor trips since 1989 related to Reactor Trip System testing. These proposed changes in test interval will provide a reduction in spurious trip rate providing a potential cost savings in excess of $100,000 over the DBNPS's remaining life. These proposed changes will also provide a reduction in resource requirements with a potential savings of $600,000 over the DBNPS's remaining life.

The proposed changes will make the following revisions to Technical Specification 3/4.3.1.1 - Reactor Protection System Instrumentation and Technical Specification 3/4.3.2.3 - Anticipatory Reactor Trip System Instrumentation: Table 4.3-1, Reactor Protection System Instrumentation Surveillance Requirements, for Functional Unit 12, Control Rod Drive Trip Breakers, and

for Functional Unit 13, Reactor Trip Module Logic, revise the channel functional test frequency from monthly on a staggered test basis to semiannually on a staggered test basis; Table 4.3-17, Anticipatory Reactor Trip System Instrumentation Surveillance Requirements, for Functional Unit 3, Output Logic, revise the channel functional test frequency from monthly to semiannually on a staggered test basis.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

These changes in the surveillance test intervals for the channel functional tests affect the control rod drive trip breakers of the Control Rod Drive Control System, and the reactor trip modules of the Reactor Protection System and the Anticipatory Reactor Trip System output logic.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

Reactor Protection System:

The purpose of the Reactor Protection System (RPS) is to initiate a reactor trip when a sensed parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. In this manner, the reactor core is protected from exceeding design limits and the Reactor Coolant System (RCS) is protected from overpressurization. The RPS monitors the following generating station variables:

1. Total out-of-core neutron flux
2. RCS coolant flow
3. RCS Pump Status
4. RCS reactor outlet temperature
5. RCS pressure
6. Containment Vessel pressure
7. Out-of-core neutron flux imbalances

The RPS is described in the DBNPS Updated Safety Analysis Report (USAR), Section 7.2. The RPS consists of four identical protection channels which are redundant and independent. Each channel is served by its own independent sensors which are physically isolated from the sensors of the other protective channels. Each sensor supplies an input signal to one or more signal processing strings in the RPS channel. Each signal processing string terminates in a bistable which electronically compares the processed signal with trip setpoints. All bistable contacts are connected in series.

In the normal untripped state, the contact associated with each bistable will be closed, thereby energizing the channel terminating relay. A trip of one of the channel bistables in one channel causes a half trip in each of the four RPS channels. The trip of a bistable in a second channel completes the two-out-of-four logic and causes a full trip of each RPS channel. The full trip of each channel deenergizes the undervoltage coils and undervoltage relays of the channel's respective Control Rod Drive (CRD) trip breaker in the Control Rod Drive Control System (CRDCS). Each RPS channel contains a reactor trip module which performs the two-out-of-four trip logic and provides the signals to open the channels' associated CRD trip breaker.

Anticipatory Reactor Trip System:

The purpose of the Anticipatory Reactor Trip System (ARTS) is to initiate a reactor trip when a sensed parameter exceeds its setpoint value indicating the approach of an unsafe condition, thereby, reducing the magnitude of pressure and temperature transients on the RCS caused by loss of feedwater events or turbine generator trips. The ARTS monitors the following generating station variables:

1. Turbine-Generator status
2. Main Feedpump Turbine status
3. Steam and Feedwater Line Rupture Control System status

The ARTS is described in the DBNPS Updated Safety Analysis Report, Section 7.4.1.4. The ARTS contains four redundant and independent channels. The turbine-generator trip input is automatically bypassed at 45% of rated thermal power or less. The other two inputs, Main Feedwater Pump Turbine (MFPT) and Steam and Feedwater Line Rupture Control System (SFRCS), are active in Mode 1. One group of pressure switches monitors the fast acting solenoids for the turbine generator stop valves and will trip the reactor when the main turbine is tripped. Another group of pressure switches monitor the oil pressure which is associated with the control valves for both MFPTs. When these sensors detect a loss of both MFPTs, the reactor will be tripped. Four additional input signals from the SFRCS will trip the reactor when the SFRCS is initiated.

Each group of four channels is connected to one of four two-out-of-four logic gates. The output from these two-out-of-four gates is applied to the associated undervoltage coils and the undervoltage relays for the CRD trip breakers in the CRDCS.

Control Rod Drive Control System:

The function of the Control Rod Drive Control System - Trip Portion is to interrupt power to the control rod drive mechanisms to insert control rods upon receipt of a RPS, ARTS, Diverse SCRAM System or manual trip signal.

A trip signal from the RPS or ARTS is applied to the undervoltage coils and the undervoltage relays of the CRD trip breakers causing the breakers to trip open, thereby removing power from the control rod drive motors resulting in the insertion of control rods and a reactor trip.

The trip portion of the CRDCS is described in the DBNPS Updated Safety Analysis Report, Section 7.4.1.1. The CRDCS trip logic is designed so that when power is removed from the control rod drive mechanisms, the roller nuts disengage from the lead screw, and a free-fall gravity insertion of the control rods occurs. Two diverse and independent trip methods, in series, are provided for removal of power to the mechanisms. First, a trip is initiated when power is interrupted to the undervoltage coils of the main A.C. feeder breakers and to the undervoltage relays in the shunt trip circuits. Second, a trip is initiated when the gating signals to the silicon controlled rectifiers (SCR) are interrupted. Since parallel power feeds are provided to the control rod drive mechanisms, interruption of

both feeds is required for trip action in either method of trip. There are two CRD trip breakers per power feed, each associated with one of four Reactor Protection System channels.

The primary method of trip interrupts power to the CRD mechanism power supplies. Power circuit breakers equipped with instantaneous undervoltage coils and shunt trip devices are used as primary trip devices. The RPS channels energize the undervoltage coil of the breakers. A trip breaker can remain closed only if its undervoltage coil is energized. Upon loss of voltage at the undervoltage coil due to interruption by an RPS, ARTS or manual trip signal, the CRD trip breaker trips open. No external power is required to trip the breakers which have stored-energy trip mechanisms. The trip breakers must be manually reset once tripped. Breaker reset is possible only after the trip signal is reset. The shunt trip undervoltage relay is installed in parallel with the undervoltage coil of the CRD trip breaker. Voltage interruption due to a trip signal deenergizes the undervoltage relay energizing the shunt trip device which is powered from essential 125 VDC, thereby tripping the CRD trip breaker.

The second trip method interrupts the gate control signals to the SCRs in each of the nine CRD mechanisms motor power supplies, and the motor return power supply. The trip devices in this case are ten relays connected with their coils in parallel. Contacts of these relays interrupt the gate control signals to the SCRs in each power supply. When the gate signals are interrupted, the SCRs will revert to their open state on the next negative half-cycle of the applied A.C. voltage, thus removing all power at the outputs of the motor power supplies. Because the power supplies have redundant halves, two sets of ten relays each are provided. RPS channel 3 energizes one set of trip relays and RPS channel 4 energizes the other set through auxiliary relays in the breakers. The trip relays can remain in their non-tripped state only if the associated RPS channel is energized. When an RPS channel trips, the associated trip relays deenergize, interrupting the SCR gate control signals.

No trip bypasses or interlocks are provided in the trip circuits.

EFFECTS ON SAFETY:

The proposed changes increase the channel functional test surveillance interval for the Reactor Protection System Control Rod Drive Trip Breakers, Reactor Trip Module Logic and the Anticipatory Reactor Trip System Output Logic to six months on a staggered test basis. Currently the Reactor Protection System Functional Units are tested monthly, on a staggered test basis, and the Anticipatory Reactor Trip System output logic is tested monthly.

Industry operating experience indicates an improvement in reactor trip breaker reliability has been achieved primarily due to upgrades installed in response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events." NRC sponsored research, reported in NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," December 1992, noted since diverse trip features were incorporated in reactor trip breakers, there have been significant improvements in

breaker reliability. Additionally, NUREG/CR-4715, prepared for the Nuclear Plant Aging Research Program, noted that a primary stress mechanism for the breaker is routine mechanical cycling associated with testing. NRC NUREG-1366 recommended owner groups consider a change in the test interval.

B&W Owners Group Topical Report BAW-10167, Supplement 3, January 1995, "Justification for Increasing The Reactor Trip System On-Line Test Interval" (BAW-10167), provides the analysis and technical justification for increasing the trip devices test interval. The DBNPS uses General Electric manufactured model AK breakers as analyzed in BAW-10167. The results of the analysis show that test interval extension is not a significant contributor to system unavailability or core damage risk. Specifically, for the DBNPS the incremental change in unavailability is shown to be 1.4E-11 failure/demand and the net incremental risk (frequency of core damage/reactor-year) is shown to be -3.2E-09. Toledo Edison has reviewed BAW-10167 and found it applicable to the DBNPS.

Toledo Edison has also reviewed the surveillance test and preventive maintenance history of the DBNPS reactor trip breakers, reactor trip module logic and Anticipatory Reactor Trip System output logic for the last eight years. Maintenance history and data trending indicates that various reactor trip breaker parameters (i.e., undervoltage device drop-out or pickup voltages, trip shaft torques, response times and insulation resistances) have been consistently found to be within the range of normal operation as a result of the existing preventive maintenance program at the DBNPS. Further, no significant changes were observed in these parameters when the preventive maintenance interval for the reactor trip breakers was increased from 12 to 18 months in 1991. The maintenance history and surveillance test results of the reactor trip module logic and the ARTS output logic also show these components have consistently met their design and operational requirements over the past eight years.

Based on the good historical performance of these components, the results of BAW-10167, the low potential for a significant increase in failure rates of the components under a longer test interval, and the introduction of no new failure modes, it is concluded there is no adverse effect on nuclear safety by increasing the test intervals to six months. Furthermore, it remains acceptable to allow the continued application of Technical Specification 4.0.2 (interval extension of 25%) when it is utilized on a non-routine basis.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(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 the proposed changes do not make a change to any accident initiator, initiating condition or assumption. The proposed changes do not involve a significant change to the plant design or operation. The proposed changes do not significantly increase system unavailability as discussed in BAW-10167.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not significantly contribute to reactor trip system unavailability or increase the risk of core damage as discussed in BAW-10167, do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation and do not provide a new radiation release path or alter potential radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety because the proposed changes do not reduce the margin to safety which exists in the present Technical Specifications or Updated Safety Analysis Report. The increase in system unavailability due to the surveillance test interval extension is insignificant as shown in BAW-10167. The Technical Specification operability requirements and action statements are not changed.

CONCLUSION:

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 concerns 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.

ATTACHMENT:

Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. Updated Safety Analysis Report Section 7.2, "Reactor Protection System (RPS)".

2. Updated Safety Analysis Report Section 7.4.1.1, "Control Rod Drive Control System (CRDCS)- Trip Portion".

3. Updated Safety Analysis Report Section 7.4.1.4, "Anticipatory Reactor Trip System (ARTS)".

4. B&W Owners Group Topical Report BAW-10167, Supplement 3, January 1995, "Justification for Increasing The Reactor Trip System On-Line Test Intervals."

5. Generic Letter 83-28, " Required Actions Based on Generic Implications of Salem ATWS Events", July 8, 1983.

6. NUREG-1366, "Improvements to Technical Specification Surveillance Requirements", December 1992.

7. NUREG/CR-4715, "An Aging Assessment of Relays and Circuit Breakers and Systems Interactions," June 1987.