APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached is a requested change to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3 Appendix A, Technical Specifications. Also included is the Safety Assessment and Significant Hazards Consideration.

The proposed change (submitted under cover letter Serial Number 1885) concerns:

Technical Specification 3/4.3.2, Instrumentation, Steam and Feedwater Rupture Control System Instrumentation.

By: _____
D. C. Shelton, Vice President - Nuclear

Sworn and subscribed before me this 19th day of December, 1990.

_____
Notary Public, State of Ohio

EVELYN L. DRESS
NOTARY PUBLIC, STATE OF OHIO
My Commission Expires July 28, 1994

The following information is provided to support issuance of the requested change to the Davis-Besse Nuclear Power Station, Unit Number 1 Operating License Number NPF-3, Appendix A, Technical Specifications, Technical Specification 3.3.2.2, Table 3.3-11.

A. Time required to implement: This change is to be implemented within 45 days after NRC issuance of the License Amendment by the NRC.

B. Reason for change (License Amendment Request Number 90-0046): This change will minimize the possibility of an inadvertent main steam low pressure trip occurring during plant cooldown and heatup by modifying the low pressure Steam and Feedwater Rupture Control System trip block permit setpoint from 700 psig to 750 psig and increasing the steam pressure where the block permit is automatically removed to 800 psig.

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

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION

FOR LICENSE AMENDMENT REQUEST NUMBER 90-0046

TITLE

Proposed change to the Davis-Besse Nuclear Power Station, Unit 1 Operating License, Appendix A, Technical Specification 3/4.3.2, Safety System Instrumentation, Steam and Feedwater Rupture Control System Instrumentation, Table 3.3-11 Steam and Feedwater Rupture Control System Instrumentation.

DESCRIPTION

The purpose of this Safety Assessment and Significant Hazards Consideration is to review the proposed change to the Davis-Besse Nuclear Power Station Unit No. 1 Technical Specifications (TS) to ensure that the change does not constitute a significant hazards consideration. The proposed TS change is to increase the Steam and Feedwater Rupture Control System (SFRCS) Main Steam (MS) low pressure block permit setpoint specified in TS Table 3.3-11 from 700 psig to 750 psig and to increase the steam pressure where the block is automatically removed from 750 psig to 800 psig. This change will increase the pressure margin between the SFRCS block permit and the SFRCS MS low pressure trip setpoints, thereby minimizing the possibility of an inadvertent MS low pressure trip during plant cooldown operations. This change will also increase the margin between the automatic block reset and the low pressure trip reset to ensure that the low pressure trips have cleared prior to the automatic reset of the SFRCS during plant startup operations.

SYSTEMS, COMPONENTS AND ACTIVITIES AFFECTED

Steam and Feedwater Rupture Control System (SFRCS)

SAFETY FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES

The SFRCS is an automatic system designed to detect and mitigate the effects of major upsets in the MS and Main Feedwater (MFW) systems, including MS and MFW line ruptures, loss of MFW events, Steam Generator (SG) overfeed, and a loss of Reactor Coolant System (RCS) forced circulation cooling. The SFRCS detects these events through sensing and logic channels and mitigates their consequences by automatically positioning valves in the MS, MFW, and Auxiliary Feedwater (AFW) systems with appropriate actuation signals dependent upon the initiating event.

The SFRCS consists of four identical sensing and logic channels housed in two electrically separate cabinets. Each cabinet consists of two redundant sensing and logic channels. Logic Channels 1 and 3 are located in Cabinet 1 and form Actuation Channel 1 (predominantly SG 1). Logic Channels 2 and 4 are located in Cabinet 2 and form Actuation Channel 2 (predominantly SG 2).

The sensing channels consist of the instrumentation used to monitor the various parameters which provide inputs to the logic channels. These inputs include:

- SG high and low water level
- MS low pressure
- SG to MFW differential pressure
- RCP high and low motor current

The only SFRCS sensing instrumentation relevant to this proposed change is the MS low pressure signal and high SG water level signal.

The High Water Level trip is not required for mitigation of any design basis accident. Therefore, this function will not be discussed.

A MS low pressure trip of an Actuation Channel of SFRCS during plant operation would be indicative of a main steam line break (MSLB). The MS low pressure trip instrumentation includes pressure switches for all four SFRCS logic channels on each MS header. A trip of single pressure switches in both Logic Channels 1 and 3 on a MS header would cause a trip of SFRCS Actuation Channel 1 whereas a trip of single pressure switches in both Logic Channels 2 and 4 on a MS header would cause a trip of SFRCS Actuation Channel 2. An SFRCS Actuation Channel trip would cause the complete isolation of the SG connected to the MS header experiencing the trip signal, the re-alignment of the affected SG's AFW pump to the opposite SG, and the initiation of AFW to the unaffected SG. It would also close the main steam line isolation valve and selected main feedwater valves on the unaffected SG.

The SFRCS also includes a manual low pressure block permissive feature that allows the operator to block the SFRCS MS low pressure and SG High Water Level trip signals during plant cooldown. This manual operator action is intended to prevent inadvertent actuation and unnecessary challenges to SFRCS and associated systems during plant cooldown. Each MS header contains four pressure switches for the block permissive signal. Two of these pressure switches are associated with a single SFRCS Logic Channel and the remaining two pressure switches on that MS header are associated with the complimentary logic channel in that Actuation Channel. This results in the four pressure switches for the block permit on the MS header for SG 1 being used to block Actuation Channel 1 and the four pressure switches on the MS header for SG 2 being used to block Actuation Channel 2. Once a block permit for a channel is received, manual action is required to actually block that channel from tripping. As required to comply with IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," for a protection system and to satisfy the Technical Specification requirements, during plant heatup the SFRCS low pressure block signal is automatically removed. This action ensures that the safety function of the SFRCS MS low pressure trip signal is activated automatically during heat-up operations and remains activated when the plant is critical in either Modes 1 or 2.

EFFECTS ON SAFETY

DBNPS Technical Specifications, Limiting Condition for Operation 3.3.2.2 requires that:

"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 and with RESPONSE TIMES as shown in Table 3.3-13."

One of the types of SFRCS sensing instrumentation listed in Table 3.3-11 is the MS low pressure instrumentation. Table 3.3-12 lists a trip setpoint for the SFRCS MS low pressure trip signal as greater than or equal to 591.6 psig. A footnote in Table 3.3-11 for the low pressure instrumentation channels states that this instrumentation:

"May be bypassed when steam pressure is below 700 psig. Bypass shall be automatically removed when the steam pressure exceeds 750 psig."

This proposed TS change will increase the steam pressure in the above footnote to 750 psig below which the SFRCS MS low pressure instrumentation can be manually bypassed. Additionally, the steam pressure above which the block permit is automatically removed will be increased to 800 psig.

As noted above, there are two block permit pressure switches for each SFRCS MS low pressure logic channel. During pressure increase, the block in both logic channels associated with an Actuation Channel needs to reset to automatically remove the block in the associated Actuation Channel. However, only one of the two pressure switches in each logic channel needs to automatically reset to remove that logic channel's block. The MS low pressure trip signal is provided by separate pressure switches which have much wider reset dead bands than the Block Permit pressure switches. This could result in the block being automatically removed prior to the trip signal having cleared causing an SFRCS activation. Since the same switches are used for the block permit and the automatic reset, both will be addressed.

The reason for the setpoints potentially overlapping as described above is that the reset dead band associated with the only available environmentally qualified replacement low pressure trip switches is larger than that of the previously installed switches.

The primary purpose of the MS low pressure instrumentation of the SFRCS is steam line break detection and mitigation in Modes 1 and 2. During plant cooldown operations, the RCS temperature closely approaches the saturation temperature associated with the SG secondary side pressure. Consequently, the present block permit value of 700 psig corresponds to an RCS temperature of approximately 506°F. Since the nominal RCS temperature in Mode 3 immediately following a reactor trip is approximately 545°F, which corresponds to the saturation temperature for the post-trip turbine bypass valve setpoint of 995 psig, SFRCS main steam line break protection is available only for a duration associated with approximately a 40°F cooldown in Mode 3.

Thus, the present block permit setpoint results in the SFRCS MS low pressure instrumentation channels being blocked over most of the RCS temperature range associated with Mode 3.

By raising the block permit setpoint to 750 psig, the RCS temperature at which the MS low pressure instrumentation can be blocked is increased from approximately 506°F to approximately 514°F. This represents an increase of 8°F in the range of RCS temperatures in Mode 3 where the MS low pressure instrumentation would be unavailable during cooldown operations. The RCS temperatures from 506°F to 514°F represent a range of transient plant operations in Mode 3 and do not represent temperatures in Mode 3 where the RCS would be stabilized for any long periods of time. Using a nominal cooldown rate of 15°F/hr, the raising of the block permit value to 750 psig would increase by approximately 30 minutes the period of time in Mode 3 where the MS low pressure instrumentation would be unavailable during a normal plant cooldown operation. The probability that a MSLB would occur in this short time period is extremely low.

By raising the automatic reset pressure of the block to 800 psig from 750 psig, the dead band associated with the resetting of the low pressure trip switches will not overlap the automatic reset of the block switches. Based on past surveillance tests of the block permit pressure switches, the switches typically reset within 20-30 psi above the block permit setpoint. Consequently, the 50 psi difference between the block permit setpoint and automatic reset point specified by Technical Specifications is considered to be an appropriate pressure range for the equipment in use.

By raising the automatic reset value to 800 psig, the RCS temperature at which the block automatically resets is increased from approximately 514°F to approximately 520°F. This increases by 6°F the range of RCS temperature in Mode 3 where the MS low pressure instrumentation would be unavailable during heat-up operations. However, it still ensures that the automatic reset occurs before the plant enters Mode 2, since per TS 3.1.1.4 the plant is not allowed to go critical until RCS T_{avg} is greater than or equal to 525°F.

Using a nominal heat-up rate of 15°F/hr, the raising of the automatic reset setpoint to 800 psig would increase the time period during heat-up where the MS low pressure instrumentation is blocked by approximately 30 minutes. As with the increased time period associated with plant cooldown operations, this time period is so short that the probability of a MSLB during this interval is extremely low.

Since during normal plant operation in Modes 1 and 2 the MS line pressure is typically 870 psig, the raising of the block permit pressure to 750 psig and the automatic reset to 800 psig has no impact upon plant operation in Modes 1 and 2. Use of the block permit in Modes 1 and 2 is not possible due to the large difference in pressure between its setpoint and the normal MS operating pressure. Consequently, the protection against MSLBs during power operation provided by the MS low pressure instrumentation is unaffected by the proposed change.

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 change and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station Unit 1 in accordance with these changes would:

1a.) Not involve a significant increase in the probability of an accident previously evaluated because the pressure switches associated with the change do not initiate any accident previously analyzed. The pressure switches only allow a manual bypass function for the MS low pressure trip switches to be activated by the operators. Additionally, the potential for an inadvertent SFRCS MS low pressure trip during plant heatup or cooldown operations will be reduced.

1b.) Not involve a significant increase in the consequences of an accident previously evaluated because the pressure switches associated with the change do not play any mitigating role in any accident previously analyzed. The pressure switches only allow a manual bypass function for the MS low pressure trip switches to be performed by the operators during controlled evolutions during Mode 3. Additionally, the potential for an inadvertent SFRCS MS low pressure trip during plant heatup or cooldown operations will be reduced. This change does not alter the radiological consequences of the bounding main steam line break accident evaluated in the USAR.

2a.) Not create the possibility of a new kind of accident from any accident previously evaluated because the setpoint change does not alter the safety function of SFRCS or any associated systems. The revised setpoints provide the same function as before and do not introduce failure modes that are not bounded by existing analyzed events.

2b.) Not create the possibility of a different kind of accident from any accident previously evaluated because the setpoint change does not alter the safety function of SFRCS or any associated systems. The revised setpoints provide the same function as before and do not introduce failure modes that are not bounded by existing analyzed events.

3.) Not involve a significant reduction in a margin of safety because the change minimizes the possibility of an unnecessary actuation of the AFW system during plant cooldown and heat-up operations. The change in the setpoints has no impact upon the availability of SFRCS during plant power operations and does not appreciably increase the time period in Mode 3 where the SFRCS main steam low pressure trip signal is blocked.

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.