APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are 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 1902) concern:

Technical Specification 3/4.1.1.3, Reactivity Controls System Technical Specification Section 6.9.1.7, Core Operating Limits Report.

For: D. C. Shelton, Vice President - Nuclear

By: TJM

T. J. Myers, Director - Technical Services

Sworn and Subscribed before me this 6th day of February, 1991.

Evelyn L. Dress

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, Section 3/4.1.1.3 and 6.9.1.7.

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

B. Reason for Change (License Amendment Request Number 90-0043):

Since a new Negative Moderator Temperature Coefficient must be calculated for each reload cycle, relocation to the Core Operating Limits Report will minimize the required number of license amendment applications submitted to the NRC for approval.

C. Safety Assessment and Significant Hazards Consideration: See Attachment 1.

D. Change to the Core Operating Limits Report: See Attachment 2.

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION

FOR LICENSE AMENDMENT REQUEST NO. 90-0043

DESCRIPTION OF PROPOSED ACTIONS

The purpose of this evaluation is to review proposed changes to Technical Specification 3.1.1.3c that will allow a moderator temperature coefficient (MTC) more negative than the current limit of -3.0×10^{-4} $\Delta k/k/^{o}F$. The new negative MTC limit will be fuel cycle-specific, thus requiring the actual value of the limit to appear in the Core Operating Limits Report (COLR) for that fuel cycle, as allowed in NRC Generic Letter 88-16. Aside from this change, the wording and intent of Technical Specification 3.1.1.3c and its associated Surveillance Requirements (4.1.1.3.1 and 4.1.1.3.2) will not change. Technical Specification 6.9.1.7 will also be modified to reflect the revised contents of the COLR.

The introduction of eighteen month fuel cycles, along with efforts to reduce the number of assemblies in each reload feed batch, forces overall core burnups to be higher at end of life conditions. By increasing core burnup, the core average plutonium concentration also increases, which has the effect of causing moderator temperature coefficients to be more negative. For Cycle 7 (startup July of 1990), the predicted MTC at rated full power conditions at end of core life was more negative than the current limit of -3.0×10^{-4} $\Delta k/k/^{o}F$. For future fuel cycles, this problem would be even more severe, and could greatly inhibit future fuel cycle planning and flexibility unless a more negative MTC limit were justified and implemented.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED

Reactor Core

SAFETY FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES

The nuclear fuel in the reactor core produces heat through the fissioning of uranium and plutonium. This heat is ultimately used to produce steam which drives the turbine to produce electricity. The safety functions performed by the reactor core and the nuclear fuel are to retain the fuel in an appropriate geometry for heat removal, and to prevent the migration of radioactive fission products away from the fuel pellets by encapsulating the pellets in Zircaloy cladding.

EFFECTS ON SAFETY

Analysis

The purpose of a negative limit for moderator temperature coefficient is to prevent large, positive reactivity insertions to the reactor core during postulated events that lead to a rapid cooldown of the Reactor Coolant System (RCS). Six of the events described in the Davis-Besse Updated Safety Analysis Report (USAR) are potentially impacted by a negative MTC at end of life (EOL) conditions. These six events are:

Event Description	USAR Section
Dropped Control Rod Assembly	15.2.3
Inactive RCS Pump Startup	15.2.6
Excessive Heat Removal Due To Feedwater System Malfunction	15.2.10
Minor Secondary Pipe Break	15.3.2
Control Rod Assembly (CRA) Ejection	15.4.3
Steam Line Break	15.4.4

Each of these events was reevaluated and will be discussed separately with respect to the proposed change in the negative MTC limit.

Dropped Control Rod Assembly Event

The dropped control rod assembly event causes sudden reductions in both neutron and thermal power, resulting in a cooldown of the RCS by as much as 20°F. This cooldown overcompensates for the inserted rod worth, increasing neutron power above the initial conditions. Generically applicable safety analysis calculations have been performed assuming an EOL hot full power (HFP) MTC of -4.0×10^{-4} $\Delta k/k/^{\circ}F$, and, although the transient response is slightly more severe than the current USAR analysis, the results of these calculations continue to meet the Safety Evaluation Criteria of USAR Section 15.2.3.2.1.

Inactive RCS Pump Startup Event

The inactive RCS pump startup event postulates the startup of two inactive reactor coolant pumps (RCPs) while the reactor is operating at 60 percent of rated full power, thus injecting cooler water from the inactive RCS loops into the core. Generically applicable safety analysis calculations have been performed assuming an EOL HFP MTC of -4.0×10^{-4} $\Delta k/k/^{\circ}F$, and, although the transient response is slightly more severe than the current USAR analysis, the results of these calculations continue to meet the Safety Evaluation Criteria of USAR Section 15.2.6.2.1. It should be noted that this evaluation is extremely conservative in that power operation with only two RCS pumps is not permitted at Davis-Besse.

Excessive Heat Removal Due To Feedwater System Malfunction Event

This event is initiated by either a sudden reduction in feedwater temperature, caused by bypassing the feedwater heaters, or by a sudden increase in feedwater flow, caused by the opening of feedwater control valves. This event may be initiated at either HFP or hot zero power (HZP) conditions, and the previous analysis for this event assumed an MTC of -3.0×10^{-4} $\Delta k/k/^{\circ}F$ at all conditions.

For the event beginning at HFP conditions, the response of the system and the consequences of the event are bounded by the HFP steam line break event, since they are essentially the same type of transient, with the steam line break being much more severe. A new acceptable MTC for steam line break at HFP conditions will be developed below, and that MTC will also ensure that the feedwater malfunction event will yield results that will continue to meet the Safety Evaluation Criteria of USAR Section 15.2.10.2.1.

A second initial condition for this event is at HZP conditions with all safety rods (groups 1 through 4) withdrawn from the core, but with the regulating rods (groups 5 through 7) fully inserted. Under these conditions, the event causes neutron power to increase to about 65 percent of rated full power and produces somewhat severe power peaking due to the inserted regulating rod configuration. This event has not been reevaluated and, therefore, it will continue to have a limiting negative MTC of -3.0×10^{-4} $\Delta k/k/^{\circ}F$ for hot zero power conditions. It will be demonstrated below that this HZP MTC value will be assured and bounded by the limiting negative MTC value at HZP conditions for the steam line break event, ensuring that this event continues to meet the Safety Evaluation Criteria of USAR Section 15.2.10.2.1. Therefore, the existing USAR analysis remains valid and bounding for this event at HZP conditions.

Minor Secondary Pipe Break Event

This event is essentially a small steam line break and will be bounded by the steam line break evaluation (see below).

Control Rod Assembly (CRA) Ejection Event

The control rod ejection transient is different from the other events in that it is initiated by a positive reactivity insertion not related to the MTC (ejection of a CRA). Although the ejected CRA event has been evaluated in the USAR with highly negative MTCs, the most severe consequences occur with more positive MTCs, not with more negative MTCs. Sensitivity studies already incorporated into the USAR show that this event has been evaluated at an MTC of -4.0×10^{-4} $\Delta k/k/^{o}F$, with results that meet the Safety Evaluation Criteria of USAR Section 15.4.3.2.1.

Steam Line Break Event

This event is unique from the others in that, while it is initiated at hot full power (HFP) conditions, the MTC value is, essentially, of concern only at HZP conditions or colder. This is because the steam line break event immediately results in a reactor trip, and, by design and in accordance with Technical Specification 3.1.1.1, the reactor will always be at least one percent shutdown at HZP conditions. What is of concern is the value of MTC at HZP conditions and colder, since that value will determine the total amount of positive reactivity that will be added to the core during the subsequent cooldown below HZP conditions.

The original steam line break analysis for Davis-Besse assumed an MTC of -3.0×10^{-4} $\Delta k/k/^{o}F$ at all moderator temperatures. A later analysis, also documented in the USAR, was performed using a reactivity-versus-moderator density curve which, while more realistic, was still very conservative. This curve yielded an average temperature coefficient (combination of Doppler and moderator temperature coefficients), over the range of temperatures between HZP and the minimum RCS temperature during the steam line break event, of -3.1×10^{-4} $\Delta k/k/^{o}F$. In other words, if the transient had been analyzed using a single temperature coefficient of -3.1×10^{-4} $\Delta k/k/^{o}F$ for all temperatures colder than HZP conditions, the same positive reactivity insertion would be produced. Therefore, this average temperature coefficient can, in fact, be shown to be a bounding value for the steam line break event at temperatures colder than HZP. When the Doppler coefficient used in the steam line break event (-1.77×10^{-5} $\Delta k/k/^{o}F$) is subtracted from this temperature coefficient, an equivalent MTC of -2.923×10^{-4} $\Delta k/k/^{o}F$ is obtained. Since this MTC value is less negative than the -3.0×10^{-4} $\Delta k/k/^{o}F$ value assumed at HZP conditions for the feedwater malfunction accident, the HZP temperature coefficient for the steam line break event, with its associated value of MTC, is a more restrictive limit that bounds the value of the HZP MTC for the feedwater malfunction event.

Since the coefficients defined above are derived from an existing USAR analysis, which has not been changed, the steam line break event will continue to meet the Safety Evaluation Criteria of USAR Section 15.4.4.2.1.

Application

Essentially, two limits have been defined above. For HFP conditions, a negative MTC of -4.0×10^{-4} $\Delta k/k/^{o}F$ has been shown to be acceptable with respect to the USAR Safety Evaluation Criteria for all of the events sensitive to a HFP MTC limit. For conditions at HZP and colder, a negative temperature coefficient of -3.1×10^{-4} $\Delta k/k/^{o}F$ has been defined for the events which are sensitive to a HZP MTC limit. However, since Technical Specification 3.1.1.3c refers to a negative MTC limit at rated full power, the HZP temperature coefficient must be related to a HFP MTC in order to provide a single value for the Technical Specification limit.

A three-dimensional full core geometry physics model, using the NOODLE analytic nodal code, was developed to transform temperature coefficients at HZP conditions, with all control rods inserted except the maximum worth stuck rod, into MTCs at HFP conditions with all rods withdrawn. It should be noted that the NOODLE code has been successfully benchmarked against measured reactivity coefficients at both HFP and HZP conditions and that the NOODLE code has been topically approved by the NRC. This transformation process, which accounts for the effects of moderator density, soluble boron concentration, control rods, and fuel temperature between HZP and HFP conditions, provides for a fuel cycle-specific transformation from the limiting HZP temperature coefficient for the steam line break event (-3.1×10^{-4} $\Delta k/k/^{o}F$) to an equivalent HFP MTC limit for the steam line break event. The new negative MTC limit, which will appear in the Core Operating Limits Report (COLR), will either be the fuel cycle-specific steam line break value described above or the -4.0×10^{-4} $\Delta k/k/^{o}F$ value used in other events, whichever is least negative.

The above-described process has been applied for the Davis-Besse Cycle 7 core. For Cycle 7, the transformed steam line break HFP MTC has a value of -3.62×10^{-4} $\Delta k/k/^{o}F$, which is less negative than -4.0×10^{-4} $\Delta k/k/^{o}F$. Therefore, the negative HFP MTC limit for the Cycle 7 core is -3.62×10^{-4} $\Delta k/k/^{o}F$. This value will appear in Table 2 of the COLR, which will be referenced by Technical Specification Limiting Condition for Operation 3.1.1.3c.

The surveillance requirements for MTC (4.1.1.3.1 and 4.1.1.3.2) will remain unchanged.

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 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 there are no accidents whose probabilities of occurrence are related to the value of the MTC.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because it has been demonstrated that all of the USAR accidents sensitive to a negative MTC still meet their USAR Safety Evaluation Criteria under the proposed new limits.

2a. Not create the possibility of a new kind of accident from any accident previously analyzed because a more negative MTC is only a concern during RCS overcooling transients that have already been addressed in the USAR and the value of the MTC cannot create a new accident.

2b. Not create the possibility of a different kind of accident from any accident previously analyzed because a more negative MTC is only a concern during RCS overcooling transients that have already been addressed in the USAR and the value of the MTC cannot create a different accident.

3. Not involve a significant reduction in a margin of safety because all events sensitive to a negative moderator temperature coefficient have been evaluated with respect to the proposed new limits in a very conservative fashion and have shown no significant change in transient response, and because the proposed change in the negative MTC limit is relatively small compared to the conservatisms in the evaluation. Further, all events sensitive to a negative MTC will continue to meet their appropriate USAR Safety Evaluation Criteria under the proposed new limits.

CONCLUSIONS

On the basis of the above, Toledo Edison has determined that this 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, the License Amendment Request does not constitute an unreviewed safety question.

ATTACHMENTS

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

RJB:tam