Enclosure 1

Revised Technical Specification Pages for the MSSV Amendment

Unit 1

Page	
3/4 7-2	Replace
3/4 7-3	Replace
B 3/4 7-1	Replace
B 3/4 7-2	Replace

Unit 2

Page	
3/4 7-2	Replace
3/4 7-3	Replace
B 3/4 7-1	Replace
B 3/4 7-2	Replace

Technical Specification Markups

Changes Marked with ***Bold, Italicized Print*** and ~~Strikethroughs~~

Table 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	~~65~~ ***48***
3	~~43~~ ***28***

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	**
2	**
3	**

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**These values left blank pending NRC approval of 2 loop operation.

TABLE 3.7-3
STEAM LINE VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ~~(±1%)~~* ***(±3%)****	ORIFICE SIZE (SQ. IN.)
a. Q1N11V0 - 10A, 11A, 12A	1075 psig	16
b. Q1N11V0 - 10B, 11B, 12B	1088 psig	16
c. Q1N11V0 - 10C, 11C, 12C	1102 psig	16
d. Q1N11V0 - 10D, 11D, 12D	1115 psig	16
e. Q1N11V0 - 10E, 11E, 12E	1129 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*****After testing, the valves will be left at ±1%.***

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~12,823,371~~ ***at least 12,984,660*** lbs/hr which is ~~110~~ ***112*** percent of the total secondary steam flow of ~~11,665,792~~ ***11,613,849*** lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are ***consistent with the assumptions used in the accident analysis***. ~~derived on the following bases:~~

~~For 3 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(V)}{X} x (109)}$$

~~For 2 loop operation~~

$$\cancel{SP = (\frac{X) - (Y)(U)}{X} x (66)}$$

~~Where:~~

~~SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER~~

~~V = Maximum number of inoperable safety valves per steam line~~

~~U = Maximum number of inoperable safety valves per operating steam line~~

~~109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation~~

~~66 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation.~~

~~X = Total relieving capacity of all safety valves per steam line in lbs/hour~~

~~Y = Maximum relieving capacity of any one safety valve in lbs/hour~~

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 330 gpm at a pressure of 1133 ~~psig~~ ***psia*** to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1133 ~~psig~~ ***psia*** to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

Table 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	87
2	~~65~~ ***48***
3	~~43~~ ***28***

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MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 2 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	**
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3	**

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**These values left blank pending NRC approval of 2 loop operation.

Changes Marked with ***Bold, Italicized Print*** and ~~Strikethroughs~~

TABLE 3.7-3
STEAM LINE VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ~~(±1%)~~* ***(±3%)*****	ORIFICE SIZE (SQ. IN.)
a. Q2N11V0 - 10A, 11A, 12A	1075 psig	16
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d. Q2N11V0 - 10D, 11D, 12D	1115 psig	16
e. Q2N11V0 - 10E, 11E, 12E	1129 psig	16

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

*****After testing, the valves will be left at ±1%.***

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1194 psig) of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~12,775,234~~ ***at least 12,984,660*** lbs/hr which is ~~110~~ ***112*** percent of the total secondary steam flow of 11,613,849 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are ***consistent with the assumptions used in the accident analysis***. ~~derived on the following bases:~~

~~For 3 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(V)}{X} \times (109)}$$

~~For 2 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (66)}$$

~~Where:~~

~~SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER~~

~~V = Maximum number of inoperable safety valves per steam line~~

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~~109 = Power Range Neutron Flux-High Trip Setpoint for 3 loop operation~~

~~66 = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for 2 loop operation.~~

~~X = Total relieving capacity of all safety valves per steam line in lbs/hour~~

~~Y = Maximum relieving capacity of any one safety valve in lbs/hour~~

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 330 gpm at a pressure of 1133 psia to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1133 psia to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 9 hours with steam discharge to the atmosphere concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

Technical Specification Pages

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*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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3/4.7.1.1 SAFETY VALVES

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Enclosure 2

Significant Hazards Evaluation

for

Main Steam Safety Valve

Technical Specification Amendment

Joseph M. Farley Nuclear Plant

Main Steam Safety Valve Technical Specification Amendments

10 CFR 50.92 EVALUATION

As required by 10 CFR 50.91 (a)(1), an analysis is provided to demonstrate that the proposed license amendment to increase the main steam safety valve (MSSV) lift setting tolerance from ±1% to ±3% and to change high nuclear flux setpoints for multiple MSSVs out of service involves no significant hazards consideration.

Proposed Change

The proposed change for Table 3.7-3 of the Farley Nuclear Plant Technical Specifications includes the revision to the MSSV setpoint tolerance from ±1% to ±3% and also modifies the bases to Technical Specification 3/4.7.1.1 to increase the relieving capacity of the MSSVs to at least 12,984,660 pounds per hour which corresponds to approximately 112% of total secondary steam flow at 100% rated thermal power. In addition, modifications to Table 3.7-1 are proposed to reduce the allowable power range neutron flux high setpoints for multiple inoperable steam generator safety valves. These changes are consistent with the current analyses for Farley Units 1 and 2. In addition, this amendment includes an editorial correction to Bases 3/4.7.1.2 which should indicate required auxiliary feedwater flow at 1133 psia rather than 1133 psig.

Background

The MSSVs provide overpressure protection for the Farley steam generators. These automatic pressure relieving devices are designed to pass at least 110% of the maximum guaranteed steam flow at a steam generator shell pressure not greater than 110% of the design pressure of the steam generator. This is the maximum pressure allowed by the code.

The current Technical Specifications contain a ±1% tolerance on the MSSV actuation setpoint. It is the intent of the proposed change to expand the tolerance of the MSSVs to ±3%. When measuring the MSSV setpoint for surveillance, the valves will be set to a ±1% measured, as-left tolerance; however, should a valve be found outside the ±1% tolerance band but within the proposed ±3% tolerance band, it is acceptable since ±3% tolerance is the new analyzed condition. Other proposed changes reflect the current analysis basis for Farley Nuclear Plant for having one or more MSSVs inoperable.

The Bases 3/4.7.1.2 correction is editorial since the Farley analyses have consistently considered auxiliary feedwater capability of 330 gpm at 1133 psia.

Analysis

The following evaluations and analyses address the effect of ±3% MSSV setpoint tolerance on mechanical aspects, design transients, non-LOCA accident analysis, LOCA and LOCA related analyses, steam generator tube rupture, radiological consequences, and emergency operating procedures.

Mechanical Aspects and Design Transients

An evaluation of the MSSVs has determined that if the MSSVs are set to within ±1% of the nominal set-pressure and later found to be outside this tolerance but within ±3%, there are no valve operational concerns. The valves will continue to function as designed.

The design transients used for the component fatigue stress analyses applicable to the Farley units have been evaluated to support an increase in MSSV setpoint tolerances to ±3%. Increasing the tolerance on these setpoints must be evaluated to confirm that no design transient is created which is more limiting than those currently applicable.

An increase of the MSSV setpoint tolerance could affect the peak steam generator pressure during the loss of load and loss of power transients that are used to define the design transients for Farley Nuclear Plant. Any increase in pressure associated with the tolerance increase will not increase the maximum postulated pressure to a value greater than the design basis of 110% of the steam generator shell design pressure. The lowest steam pressure remains above the design opening pressure for the steam generator PORVs. Adequate margin exists in the analytical assumptions for the current design transients such that no change to the component fatigue analyses is required. The current design transients remain valid for all applications.

Non-LOCA Accident Analysis

The non-LOCA accident analyses can be placed into two categories with respect to the proposed increase in MSSV setpoint tolerance to ±3%. These are (1) non-limiting transients or transients which do not actuate MSSVs, or (2) limiting transients which may actuate MSSVs.

The first set of analyses, i.e., non-limiting transients or transients which do not actuate MSSVs, include uncontrolled RCCA bank withdrawal from a subcritical condition, main steamline break, main steam line break mass and energy releases (inside and outside containment), RCCA misalignment, uncontrolled boron dilution, loss of flow/locked rotor analyses, startup of an inactive loop, feedwater system malfunction, excessive load increase, accidental depressurization of the RCS, inadvertent loading and operation of a fuel assembly in an improper position, single RCCA withdrawal at full power, and RCCA ejection. This group of transients is unaffected by the increase in the MSSV setpoint tolerance and therefore all FSAR conclusions remain valid.

The second set of analyses, i.e., limiting transients that may actuate MSSVs include uncontrolled RCCA bank withdrawal at power, loss of external electrical load and/or turbine trip, loss of normal feedwater and loss of non-emergency AC power to plant auxiliaries, inadvertent operation of ECCS during power operation, and major rupture of a main feedwater pipe. For all of these events, the current analysis of record has accounted for the increased tolerance on MSSVs and all acceptance criteria continue to be met.

Additional analyses were performed to determine the maximum allowable power range neutron flux high setpoints with inoperable main steam safety valves. The setpoint (87%) for 1 valve/loop remains the same. The setpoint for 2 valves/loop is modified to 48%, and the setpoint for 3 valves/loop is modified to 28%. These high nuclear flux setpoints reflect the $\pm$ 3% MSSV setpoint tolerance and appropriate accumulation.

LOCA and LOCA Related Evaluations

Large Break LOCA

The current large break LOCA analyses for J. M. Farley Units 1 and 2 were performed with the NRC approved Evaluation Model using BASH. After a postulated large break LOCA occurs, the heat transfer between the reactor coolant system (RCS) and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary pressure increases and the MSSVs may actuate to limit pressure. However, this does not occur in the large break evaluation model since no credit is taken for auxiliary feedwater actuation. Consequently, the secondary system acts as a heat source in the postulated large break LOCA transient and the secondary pressure does not increase. Since the secondary system pressure does not increase, it is not necessary to model the MSSV setpoint in the large break evaluation model. Therefore, an increase in the allowable MSSV setpoint tolerance for J. M. Farley Units 1 and 2 will not impact the current FSAR large break LOCA analyses.

Small Break LOCA

The small break LOCA analyses for J. M. Farley Units 1 and 2 were performed with the NRC approved Evaluation Model using the NOTRUMP code. After a postulated small break LOCA occurs, the heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases which leads to steam relief via the MSSVs. In the small break LOCA, the secondary flow aids in the reduction of RCS pressure. The current licensing basis small break LOCA analysis for J. M. Farley Nuclear Plant was performed using a MSSV setpoint tolerance of ±3%. Therefore, the increased setpoint tolerance has already been included in the analysis of record and all acceptance criteria continue to be met.

LOCA Related Events

Post-LOCA cooling, hot leg switchover to prevent potential boron precipitation, and LOCA hydraulic forcing functions have been reviewed and it has been determined that the MSSV setpoint tolerance increase has no effect on the results or conclusions for these events.

Steam Generator Tube Rupture (SGTR)

A design basis failure of a single steam generator tube was evaluated using the assumptions which were utilized in the FSAR SGTR analysis. An SGTR results in a loss of coolant inventory, and reactor trip and safety injection (SI) are assumed to occur on a low pressurizer pressure signal. After reactor trip and SI actuation, it is assumed that the secondary side pressure stabilizes at the MSSV setpoint minus the tolerance (3%). It is assumed that the RCS pressure stabilizes at the equilibrium value where the incoming SI flow rate balances the tube rupture break flow rate, which is dependent on the primary to secondary side pressure differential. The resultant equilibrium break flow rate is assumed to persist from the time of reactor trip and SI actuation until 30 minutes after the accident. A maximum SI flowrate is conservatively assumed for the design basis SGTR analysis in order to maximize the break flow.

The results of the thermal and hydraulic evaluation of the SGTR for the increased MSSV tolerance indicate that the primary to secondary break flow of 137,811 pounds would increase by 0.9% to 139,100 pounds. The amount of steam released is increased by 1.6% to 65,500 pounds. These results were used in calculating the effect of the increased MSSV tolerance on the offsite radiological consequences.

The radiological consequences were reanalyzed using the results of the previously noted thermal and hydraulic evaluations. The calculated doses remain with the NRC acceptance criteria of 10CFR100.

Radiological Consequences

The radiological consequences for FSAR analyses for loss of load/turbine trip, loss of offsite power, locked rotor, main steam line break and rod ejection remain bounding for the proposed MSSV tolerance increase.

EOPs

Emergency Operating Procedures, which could be affected by the proposed change, have been reviewed and no changes in any setpoints are required.

10CFR50.92 Conclusions

Conformance of the proposed amendment to the standards for a determination of a No Significant Hazards as performed in 10CFR50.92 is shown in the following:

1. The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

These proposed changes to the Farley Technical Specifications do not result in a condition where the design, material and construction standards of the MSSVs that were applicable prior to the proposed change are altered. The valves will continue to function as designed. All applicable safety analyses have been reviewed, evaluated or reanalyzed and all applicable safety criteria continue to be met. No accident sequences are altered because of the proposed amendment. The radiological consequences for the Steam Generator Tube Rupture were reanalyzed and 10CFR100 criteria continue to be met. All other FSAR radiological analyses remain bounding. Analyses have been performed to justify the proposed high nuclear flux setpoint changes. All acceptance criteria for these analyses continue to be met. Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendment does not create the possibility of a new or different accident from any accident previously evaluated.

The MSSVs continue to have the required pressure relieving capacity to ensure that system design pressure remains below 110% of shell design pressure. The proposed changes are not accident initiators nor do they create any new accident scenarios or any new limiting single failures. The ability of the MSSVs to respond to an accident condition is not impaired by the proposed changes. The proposed high nuclear flux setpoints for multiple valves out of service ensure all applicable safety criteria for accident analyses are met. No new accident scenarios are created by these proposed changes. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in the margin of safety.

Acceptance criteria for accident analysis continue to be met. Radiological consequences for the affected Chapter 15 analysis remain within 10CFR100 acceptance criteria. No safety limits or safety system setpoint requires modification due to the proposed changes. The current secondary side over-pressure limit of 110% of steam generator shell design pressure is not violated. Analysis for the high nuclear flux setpoints have verified that there is no reduction in margin for the events analyzed. Therefore, there is not significant reduction in the margin of safety.

Enclosure 3

Safety Analysis

for

Main Steam Safety Valve Technical Specification Amendment

Joseph M. Farley Nuclear Plant
Main Steam Safety Valves
Safety Analysis

EXECUTIVE SUMMARY

Southern Nuclear is proposing a change to the Main Steam Safety Valve (MSSV) setpoint tolerance from ± 1% to ± 3%. If the as-found set-pressure of a MSSV is found to be within the tolerance of 3%, the measured, as-left set-pressure will be reset to be within ± 1%. The analyses indicate that the change does not adversely affect the performance of the MSSV and acceptance criteria for all accidents continue to be met. In addition, analyses to support changes to high nuclear flux setpoints for multiple MSSVs out of service were performed. Acceptance criteria for these cases was also met.

BACKGROUND

The MSSVs provide overpressure protection for the Farley steam generators. These automatic pressure relieving devices are designed to pass at least 110% of the maximum guaranteed steam flow at a steam generator shell pressure not greater than 110% of the design pressure of the steam generator. This is the maximum design pressure allowed by the applicable code.

The Technical Specifications' setpoint tolerance of ±1% was based on an ASME Section III requirement. To allow increased flexibility, a Technical Specification setpoint tolerance change to ±3% is proposed. Prior to implementing such a change, the applicable safety analyses have to be reviewed to show that the plant would not be overpressurized if a valve(s) is determined to have a set-pressure within the ± 3% tolerance. All analyses have been reviewed and all applicable criteria continue to be met.

Westinghouse identified in Nuclear Safety Advisory Letter NSAL-94-001 a deficiency in the basis for Technical Specification 3.7.1.1. This Technical Specification allows the plant to operate at a reduced power level with a reduced number of operable Main Steam Safety Valves (MSSVs). The deficiency is in the assumption that the maximum allowable initial power level is a linear function of the available MSSV capacity. The linear function is identified in the bases section for this Technical Specification.

One of the options presented in NSAL-94-001 is to perform plant-specific analyses of the Loss of Load/Turbine Trip (LOL/TT) event to analytically determine the maximum allowed power level for a given number of inoperable MSSVs. Depending on key specific plant parameters, these analyses may be able to justify the continued validity of the current Technical Specification. A plant-specific analysis has been performed for Farley Nuclear Plant.

EVALUATION

Mechanical

The MSSVs at Farley Units 1 and 2 are Dresser Model 3707R. The valve evaluation has shown that if the MSSVs are set to within ± 1% of the nominal set-pressure and later found to be outside this tolerance but within ± 3%, there are no ASME valve operational concerns. If a valve is found to have a set-pressure outside the 1% tolerance, it should be reset to within ±1% to minimize the potential for valve leakage and set-pressure overlap problems.

The MSSVs design basis set-pressures are staggered so that they open at different pressures. The first valve set-pressure is 1075 psig, which corresponds to the steam generator shell design pressure minus the pressure loss from the steam generator to the valve. Each of the remaining valves is set at a higher pressure, such that all valves are open and at full relief without exceeding 110% of the steam generator shell design pressure. The design steam pressure for the steam generator is 1085 psig.

Design Transients

The design transient curves used for the component fatigue stress analyses applicable to the Farley units have been evaluated to support an increase in MSSV setpoint tolerances to ±3%. Increasing the tolerance on these setpoints must be evaluated to confirm that no design transient is created which is more limiting than those currently applicable.

An increase of the MSSV setpoint tolerance could affect the peak steam generator pressure during the loss of load and loss of power transients that are used to define the design transients for the Farley Nuclear Plant. Any increase in pressure associated with the tolerance increase will not increase the maximum postulated pressure to a value greater than the design basis of 110% of the steam generator shell design pressure. The lowest steam pressure remains above the design opening pressure for the steam generator PORVs. Adequate margin exists in the analytical assumptions for the current design transients such that no change to the component fatigue analyses is required. The current design transients remain valid for all applications.

Non-LOCA

Currently the Farley Unit 1 and 2 Technical Specifications require that the main steam safety valves (MSSV) be verified to be operable within a ± 1% tolerance of the corresponding nominal lift set-pressure. The analyses and evaluations for non-LOCA transient documented within demonstrate that the MSSV setpoint tolerance can be increased to ±3%.

Evaluation

The following evaluation discusses how the MSSVs have been historically modeled and how the safety valves are currently modeled in the non-LOCA safety analyses. This is followed by a discussion of how the increase in the MSSV tolerance has been incorporated into the non-LOCA safety analyses.

Historic Safety Analyses Safety Valve Model

Historically, the ±1% tolerance of the MSSVs has not been explicitly modeled in the non-LOCA safety analyses. Rather, the safety analyses assumed that the safety valves were open when the valves reach a pressure 3% above the nominal set-pressure. The 3% value is the point where the valves had accumulated such that the valves were relieving at their full capacity. No relief was assumed prior to reaching the accumulation pressure. In addition, the safety valves were modeled at one pressure, corresponding to the steam generator design pressure plus 3% for accumulation. This approach was acceptable because the transients presented in the FSAR only required the opening of the safety valves for a post-reactor trip condition in which the total stored energy and decay heat level would be less than the relief capacity of one bank of safety valves. The MSSVs "closed" when the pressure dropped below the accumulation pressure.

Justification of the Increased Safety Valve Tolerance

To justify an increase in the MSSV tolerance, it is assumed that the MSSVs open at the set-pressure plus 3% accumulation plus 3% tolerance. The safety analyses assume that the valves only relieve enough steam to maintain pressure at the set-pressure. It is also assumed that the secondary pressure will drop well below the opening pressure of the valve before the valve will close. Therefore, the "average" secondary pressure during the time when the valves are open will be less than the set-pressure plus the tolerance.

Two MSSV models are used in the safety analyses. The first model assumes that the safety valves open at a pressure greater than the lowest set-pressure plus 3% accumulation plus 3% tolerance. The second model assumes staggered MSSV banks at nominal set-pressures plus an accumulation allowance plus 3% tolerance. The second MSSV model models the actual operation of the MSSVs more accurately. Both MSSV models support an increase in the safety valve tolerance to ±3%.

The following section addresses the individual non-LOCA accidents with respect to the MSSV tolerance, as discussed above.

Non-LOCA Safety Analyses Event Discussion

Steamline Break Mass and Energy Releases (FSAR 6.2.1.3.11)

The steamline break mass and energy releases are not affected because the transient results in a decrease in the secondary pressure. Therefore, the MSSVs are not challenged and the mass and energy releases remain valid.

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition (FSAR 15.2.1)

The uncontrolled RCCA withdrawal from a subcritical condition event is analyzed to demonstrate that the DNB design basis is met. Since this is a rapid reactivity excursion event that is promptly terminated by a reactor trip at 35% rate thermal power, the peak thermal power is much less than that associated with full power operation (typically, less than 50% rated thermal power). This event is not limiting with respect to secondary side pressure. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Uncontrolled RCCA Bank Withdrawal at Power (FSAR 15.2.2)

The uncontrolled RCCA withdrawal at power event is analyzed to demonstrate that the DNB design basis is met. The MSSVs are modeled at a pressure greater than the lowest set-pressure plus 3% accumulation plus 3% tolerance. The results show that the peak secondary pressure is not exceeded, the DNB design basis is met and the pressurizer does not overfill for the Farley plant. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

RCCA Misalignment (FSAR 15.2.3)

The dropped RCCA(s) event is analyzed to demonstrate that the DNB design basis is met. Changes to the MSSVs tolerance do not impact the event.

Uncontrolled Boron Dilution (FSAR 15.2.4)

The chemical and volume control system malfunction that results in a decrease in the boron concentration in the RCS event is analyzed to demonstrate that sufficient operator action time is available to terminate the event prior to losing shutdown margin. The event does not examine secondary pressure since secondary pressure is bounded by other Condition II events. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Partial Loss of Forced Reactor Coolant Flow (FSAR 15.2.5)

The partial loss of reactor coolant flow event is analyzed to demonstrate that the DNB design basis is met. The MSSVs are not challenged during this transient. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Startup of an Inactive Reactor Coolant Loop FSAR (15.2.6)

The startup of an inactive reactor coolant loop event is analyzed to demonstrate that the DNB design basis is satisfied. The analysis of the event results in a very slight increase in the RCS pressure and secondary system pressure never approaches the MSSV set-pressure. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Loss of External Electrical Load and/or Turbine Trip (FSAR 15.2.7)

The loss of external electrical load/turbine trip event is analyzed with MSSVs at a pressure greater than the lowest set-pressure plus 3% accumulation plus 3% tolerance. All four cases presented in the FSAR are analyzed, i.e., the minimum feedback cases with and without pressure control and the maximum feedback cases with and without pressure control. For the minimum feedback cases, a 0 pcm/°F moderator temperature coefficient (MTC) is assumed instead of the +7 MTC presented in the FSAR. This is justified since the event is analyzed at full power conditions and the plant must have a 0 pcm/°F (or negative) MTC at full power. The analysis with this combination bounds any combination of lower power level with a positive MTC (based on the Technical Specification's MTC requirement that ramps from +7 pcm/°F at 70% rated thermal power to 0 pcm/°F at full power). The results of the analysis show that the secondary pressure limit is not exceeded. Based on the analyses, an increase on the MSSV tolerance to ±3% can be supported and the conclusions of the FSAR remain valid.

Loss of Normal Feedwater events and Loss of Non-Emergency AC Power to the Plant Auxiliaries (FSAR 15.2.8 and 15.2.9)

The loss of normal feedwater event and loss of non-emergency AC power to the plant auxiliaries are analyzed to demonstrate that the pressurizer does not become water solid. This ensures that these Condition II transients do not create a more limiting event (i.e., Condition III or IV event) due to water relief out the safety valves. With respect to DNB, these events are bounded by the loss of flow event and loss of load/turbine trip events. For these events, the staggered MSSV banks are modeled with each bank at its nominal set-pressure plus 3% accumulation plus 3% tolerance. The staggered MSSV bank model reduced the peak steam generator pressure for the loss of normal feedwater and loss of non-emergency AC power events. Therefore, the MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Excessive Heat Removal Due to Feedwater System Malfunctions (FSAR 15.2.10)

The feedwater malfunction event is unaffected by any changes to the MSSV tolerance because this event results in a decrease in the secondary pressure. Therefore, the MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Excessive Load Increase Incident (FSAR 15.2.11)

The excessive load increase event is unaffected by any changes to the MSSV tolerance because this event results in a decrease in the secondary pressure. Therefore, the MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Accidental Depressurization of the RCS (FSAR 15.2.12)

The inadvertent opening of a pressurizer safety or relief valve event results in a depressurization of RCS. The secondary pressure rises for this event; however, the MSSVs are not challenged. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Accidental Depressurization of the Main Steam System (FSAR 15.2.13)

The inadvertent opening of a safety valve, relief valve or steam dump in the main steam system results in a depressurization of main steam system. This event results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The MSSVs are not challenged for this event. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Inadvertent Operation of the Emergency Core Cooling System During Power Operation (FSAR 15.2.14)

The inadvertent operation of the ECCS during power operation event is analyzed to support a non-conservatism identified in the current licensing basis analysis. The event is analyzed to demonstrate that sufficient operator action time exists to terminate ECCS flow following a spurious safety injection signal to prevent water relief out the pressurizer safety valves. This ensures that this Condition II transient does not create a more limiting event (i.e., Condition III or IV event) due to water relief out the safety valves. For this transient, the staggered MSSV banks are modeled with each bank at its nominal set-pressure plus 3% accumulation plus 3% tolerance. The results show that the pressurizer does not fill before the operator can terminate the ECCS flow. Peak secondary pressure is not exceeded for the transient. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (FSAR 15.3.3)

The inadvertent loading and operation of a fuel assembly in an improper position does not result in any RCS transient, rather it results in peaking factor concerns. Therefore, it is not affected by the proposed change in the MSSV tolerance. The MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Complete Loss of Forced Reactor Coolant Flow (FSAR 15.3.4)

The complete loss of reactor coolant flow event is analyzed to demonstrate that the DNB design basis is met. The MSSVs are not challenged during this transient. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Single RCCA Withdrawal at Full Power (FSAR 15.3.6)

The single rod withdrawal at full power event is a DNB event. Changes to the MSSV tolerance do not impact the event.

Rupture of Main Steam Line (FSAR 15.4.2.1)

The steam line break event is unaffected by any changes to the MSSV tolerance because this event results in a decrease in the secondary pressure. Therefore, the MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Major Rupture of a Main Feedwater Pipe (FSAR 15.4.2.2)

The feedline break transient is analyzed to demonstrate that the core remains in a coolable geometry. This is ensured by demonstrating that no hot leg boiling occurs prior to the time that the auxiliary feedwater system heat removal capability exceeds the stored energy, decay heat and RCS pump heat (for the case with offsite power) levels. For this event, the staggered MSSV banks are modeled with each bank at its nominal set-pressure plus 3% accumulation plus 3% tolerance. The results indicate that no hot leg boiling occurs prior to event turn around and peak secondary pressure is not exceeded. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Reactor Coolant Pump Shaft Break (FSAR 15.4.4)

The reactor coolant pump shaft seizure and shaft break are analyzed as one event. It is assumed that when RCS flow in the faulted loop is positive, the rotor is locked, and when RCS flow in the faulted loop is negative, the rotor is free to spin (shaft break). The MSSVs are not challenged during this transient. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (FSAR 15.4.6)

The rod ejection accident is analyzed to demonstrate that applicable Condition IV criteria are satisfied. The analysis presented in the FSAR examines the peak clad temperature, fuel enthalpies, fuel melting and zirc-water reactions at the "hot spot" in the core. The MSSVs are not modeled as a part of the RCS overpressure analysis since the heat addition and overpressurization occur so rapidly. However, should the MSSVs activate following a reactor trip, adequate relief capacity exists to prevent overpressurization of the secondary side. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Steam Line Break Mass and Energy Releases - Outside Containment

The steamline break mass and energy releases outside containment are not affected because the transient results in a decrease in the secondary pressure. The MSSVs are not challenged during this transient. Therefore, an MSSV tolerance of ±3% can be supported and the conclusions of the FSAR remain valid.

Evaluation for Inoperable MSSV(s)

Introduction

Westinghouse identified in Nuclear Safety Advisory Letter NSAL-94-001 a deficiency in the basis for Technical Specification 3.7.1.1. The Technical Specification allows the plant to operate at a reduced power level with a reduced number of operable MSSVs. The deficiency is in the assumption that the maximum allowable initial power level is a linear function of the available MSSV capacity. The linear function is identified in the bases section for this Technical Specification.

One of the options presented in NSAL-94-001 is to perform plant-specific analyses of the Loss of Load/Turbine Trip (LOL/TT) event to analytically determine the maximum allowed power level for a given number of inoperable MSSVs. Depending on key specific plant parameters, these analyses may be able to justify the continued validity of the current Technical Specification. A plant-specific analysis has be performed for Farley Nuclear Plant. This evaluation provides the results of this effort.

Evaluation

Technical Specification Table 3.7-1 indicates the maximum power level at which a plant can operate, based on a reduction in the high neutron flux reactor trip setpoint, for 1, 2, or 3 inoperable MSSVs on any loop. The most limiting transient for this condition for the Farley units is the LOL/TT event presented in Section 15.2.7 of the FSAR since this transient results in the greatest demand on the relief capacity of the MSSVs. Therefore, the maximum power level at which the plant may safely operate, with inoperable MSSVs, may be determined by demonstrating that a LOL/TT incident initiated from the assumed power level, will not result in the secondary steam pressure exceeding 110% of the design pressure. For Farley Nuclear Plant, the pressure limit is 1208.5 psia {(1085 psig * 1.1) + 15 psi}.

The analysis supporting revisions to Table 3.7-1 uses a more detailed MSSV relief model compared to the lump model historically employed. The Farley units each have 5 MSSVs per loop, or 5 banks. For each bank, the analysis conservatively models steam relief to begin at the valve set-pressure plus 3% of the set-pressure (tolerance) plus 20 psi due to the pressure drop from the steam generator to the location of the MSSVs on the main steamlines. The latter is to account for the difference between the actual bank location and the analysis model which uses the steam generator pressure as the reference pressure for MSSV bank actuation. Valve accumulation is also typically modeled by a linear ramp in flow from the initial valve opening pressure to full flow 3% above the opening pressure. However, when considering a 3% tolerance, 3% accumulation, and 20 psi pressure drop, full valve flow will not be attained in banks 4 and 5 prior to exceeding the analysis pressure limit. For banks 4 and 5 a revised accumulation requirement is modeled. Bank 4 is assumed to be fully accumulated over a 2% (of the valve set-pressure) ramp and bank 5 is assumed to fully accumulate over a 10 psi ramp. This modeling is consistent with MSSV test results (Reference A) for actuation at higher pressures.

Multiple cases of the LOL/TT event, initiated from less than full power, and assuming either 1, 2 or 3 inoperable MSSVs per loop have been analyzed. The moderator temperature coefficient (MTC) was also varied according to the Technical Specification requirement (+7 pcm/°F $\leq$ 70% rated thermal power and ramping to 0 pcm/°F at 100% rated thermal power) to bound the allowed operation conditions of the plant. The LOL/TT analysis credits reactor trip signals on high neutron flux (HNF), OTΔT, and Low-Low Steam Generator Level (LLSGL). In general, when the MTC is greater than +3 pcm/° F, the power level rises until the assumed HNF setpoint is reached. This occurs relatively early in the transient. For values of MTC less than +3 pcm/°F, a reactor trip signal is generated by either the OTΔT or LLSGL functions. Note the OTΔT function provides a reactor trip at power levels supported with only 1 inoperable MSSV per loop. At the further reduced power levels required with 2 or 3 inoperable MSSVs per loop, an OTΔT signal may not be generated and protection for the lower MTC cases (<+3) is provided by the LLSGL function.

The analysis assumes a HNF setpoint equal to the proposed Technical Specification limit plus 9% uncertainty and an initial power level equal to the proposed Technical Specification HNF setpoint. The reduced power LOL/TT analysis, performed with 1, 2, or 3 inoperable MSSVs supports the following revised HNF setpoints for Technical Specification Table 3.7-1 by demonstrating that the peak steam generator and RCS pressures do not exceed 110% of their respective design values, the pressurizer will not reach a water solid condition, and the DNBR limit is not violated.

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	87
2	48
3	28

This evaluation, consistent with the Technical Specifications, does not support operation with 4 or more inoperable MSSV on any single loop.

Conclusions

A plant-specific analysis of the LOL/TT event has been performed for the Farley units to determine acceptable maximum reduced power levels for operation with 1, 2, or 3 inoperable MSSVs per loop. The LOL/TT event is the limiting transient for Farley Units 1 and 2 with respect to MSSV relief capacity and over-pressurization of the secondary steam system. The analysis demonstrates that operation at or below power levels governed by the above reduced HNF setpoints will not result in exceeding the safety analysis criteria should a LOL/TT event be initiated from these conditions.

Non-LOCA Conclusions

The analyses and evaluations documented within demonstrate that with respect to the non-LOCA safety analyses, an MSSV tolerance of ±3% and modification to the high nuclear flux setpoints with MSSVs out of service can be supported for the Farley Units.

LOCA and LOCA Related Evaluations

Large Break LOCA

The current large break LOCA analyses for J. M. Farley Units 1 and 2 were performed with the NRC approved Evaluation Model using BASH. After a postulated large break LOCA occurs, the heat transfer between the reactor coolant system (Reactor Coolant System) and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases and the MSSVs may actuate to limit the pressure. However, this does not occur in the large break evaluation model since no credit is taken for auxiliary feedwater actuation. Consequently, the secondary system acts as a heat source in the postulated large break LOCA transient and the secondary pressure does not increase. Since the secondary system pressure does not increase, it is not necessary to model the MSSV setpoint in the large break evaluation model. Therefore, an increase in the allowable MSSV setpoint tolerance for Farley Units 1 and 2 will not impact the current FSAR large break LOCA analyses.

Small Break LOCA

The small break LOCA analyses for Farley Units 1 and 2 were performed with the NRC approved Evaluation Model using the NOTRUMP code. After a postulated small break LOCA occurs, the heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, the secondary system pressure increases which leads to steam relief via the MSSVs. In the small break LOCA, the secondary flow aids in the reduction of RCS pressure. The current licensing-basis small break LOCA analysis for Farley was performed using a MSSV setpoint tolerance of 3%. Therefore, the increase has already been included in the analysis.

Post-LOCA Long Term Core Cooling

The Westinghouse licensing position for satisfying the requirements of 10 CFR 50.46 Paragraph (b), Item (5), "Long Term Cooling," concludes that the reactor will remain shut down by borated ECCS water residing in the RCS/sump after a LOCA. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a boron concentration that, when mixed with other water sources, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady state conditions at the initiation of a LOCA and considers sources of both borated and unborated fluid in the post-LOCA containment sump. The steady state conditions are obtained from the large break LOCA analysis which, as stated above, does not take credit for MSSV actuation. Thus the post-LOCA long-term core cooling evaluation is independent of the MSSV setpoint tolerance, and there will be no change in the calculated RCS/sump boron concentration after a postulated LOCA for Farley Units 1 and 2.

Hot Leg Switchover to Prevent Potential Boron Precipitation

Post-LOCA hot leg recirculation time is determined for inclusion in emergency operating procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, and the RCS, RWST, and accumulator water volumes and their associated boron concentrations. The proposed ±3% increased MSSV setpoint tolerance does not affect either the power level or the boron concentrations assumed for the RCS, RWST and accumulator in the hot leg switchover calculation for J. M. Farley Units 1 and 2.

LOCA Hydraulic Forcing Functions

The peak hydraulic forcing functions on the reactor vessel and internals occur very early in the large break LOCA transient. Typically, the peak forcing functions occur between 10 and 50 milliseconds (0.01 and 0.05 seconds) and have subsided well before 500 milliseconds (0.50 seconds). Any change in time associated with an increased MSSV setpoint tolerance would occur several seconds into the transient. Since the LOCA hydraulic forcing functions have peaked and subsided before the time at which the MSSV may actuate, the increase in the MSSV setpoint tolerance to 3% will not impact the LOCA hydraulic forcing functions calculation for Farley Units 1 and 2.

LOCA Conclusions

The effect of increasing the MSSV setpoint tolerance to ±3% for J. M. Farley Units 1 and 2 has been evaluated for each of the LOCA related analyses addressed in the FSAR. It was shown that the ±3% MSSV setpoint tolerance does not result in any design or regulatory limit being exceeded for operation. Therefore, with respect to the LOCA analyses, it can be concluded that increasing the MSSV setpoint tolerance to ±3% for J. M. Farley Units 1 and 2 will be acceptable from the standpoint of the FSAR accident analysis discussed in the safety evaluation.

EOPs

The impact of increasing the Technical Specification allowable drift tolerance to ± 3% on the MSSVs was evaluated with respect to the Emergency Operating Procedures (EOPs) since lift setpoints for the MSSVs are utilized as EOP setpoints. It is noted that although this change will increase the allowable drift of the MSSVs, the valves will continue to be set at a ±1% measured, as-left tolerance. This evaluation was based on the setpoint requirements (as footnoted) of the Westinghouse Owners Group Emergency Response Guidelines.

The following EOP setpoints which utilize MSSV lift setpoints were reviewed and evaluated:

1. Steam generator pressure for highest steamline safety valve;

2. Steam generator pressure for lowest steamline safety valve;

3. Setpoint for steam generator PORV controller (typically 25 psig below lowest safety valve set-pressure);

4. Setpoint for steam generator PORV controller (typically 25 psig below lowest safety valve set-pressure), minus 25 psi; and

5. RCP trip parameter and setpoint.

For the steam generator pressure values corresponding to the highest and lowest steamline safety valves, the MSSV set-pressures are used on the Heat Sink Critical Safety Function Status Tree to determine the appropriate procedure for implementation. The nominal value for the highest steamline safety valve setpoint is used to determine if a steam generator overpressure condition exists while the nominal value for the lowest steamline safety valve setpoint is used to identify a loss of normal steam release capability. For each of these setpoints, the increase in the allowable setpoint drift tolerances ($\pm$ 3%) between the EOP MSSV setpoints and the MSSV in-plant lift pressures will not impact the use of the setpoints in the EOPs. If the set-pressures are within 3%, no modifications to the setpoints used in the EOPs will be required.

The setpoint of the steam generator PORV controller is used in the E-3, Steam Generator Tube Rupture guideline to isolate flow from the ruptured steam generators. The 25 psi margin below the lowest safety valve set-pressure is a typical generic value that is low enough to allow for the opening of the steam generator PORV prior to lifting the safety valve and high enough to stay above the no-load steam generator pressure value in order to minimize atmospheric releases from the ruptured steam generator. Even if the MSSV lift setpoint drifts to the new allowable limit from the $\pm$ 1% tolerance to which they are set, the actual Farley EOP setpoint value for the steam generator PORV controller will ensure that the steam generator PORV will be opened prior to lifting the MSSV. Therefore, an increase in the MSSV setpoint drift tolerance from $\pm$ 1% to $\pm$ 3% will not impact this setpoint.

The setpoint for the steam generator PORV controller minus 25 psi is used in the steam generator tube rupture recovery guidelines as a setpoint for stopping feed flow to a ruptured steam generator as the steam bubble is compressed. The 25 psi margin in addition to the setpoint discussed above allows for operator action without opening the steam generator PORV. With these operating margins, an increase in the MSSV setpoint drift tolerance from $\pm$ 1% to $\pm$ 3% will not impact this setpoint.

For the RCP trip parameter and setpoint, the lowest MSSV lift pressure is used as input for the determination of when to trip the RCPs in the EOPs based on RCS pressure. This determination is conservative since a generic 3% partial open and tolerance error is assumed and instrument uncertainties are taken into account. With this conservatism and a small difference between the MSSV pressure to determine the RCP trip setpoint and an in-plant first lift pressure of less than 3%, there is no impact on the EOPs in this area.

In summary, the revision to the Technical Specification allowable drift tolerance of $\pm$ 3% on the MSSV lift setpoints will not effect the EOP setpoints listed above. The current EOP setpoints remain valid.

SGTR

Steam Generator Tube Rupture Evaluation

An evaluation of the steam generator tube rupture (SGTR) has been performed to determine the effect of the increase in MSSV tolerance.

The SGTR analysis performed for Farley Units 1 and 2, presented in Section 15.4 of the FSAR, was performed to ensure that the offsite radiation doses remain below ("a small fraction") the limits defined in 10CFR100. The primary thermal and hydraulic parameters which affect the calculation of the offsite radiation doses for an SGTR are the amount of radioactivity assumed to be available in the reactor coolant, the amount of reactor coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, and the amount of steam released from the ruptured steam generator to the atmosphere.

For the FSAR SGTR analysis, the activity in the reactor coolant is based on an assumption of 1% defective fuel, and this assumption will not be affected by the increase in MSSV tolerance. Thus, an evaluation was performed to determine the effect of the increased MSSV tolerance on the primary to secondary break flow and the amount of steam released to the atmosphere.

A design basis failure of a single steam generator tube was evaluated using the assumptions which were utilized in the FSAR SGTR analysis. An SGTR results in a loss of coolant inventory, and reactor trip and safety injection (SI) are assumed to occur on a low pressurizer pressure signal. After reactor trip and SI actuation, it is assumed that the secondary side pressure stabilizes at the MSSV setpoint minus the tolerance (3%). It is assumed that the RCS pressure stabilizes at the equilibrium value where the incoming SI flow rate balances the tube rupture break flow rate, which is dependent on the primary to secondary side pressure differential. The resultant equilibrium break flow rate is assumed to persist from the time of reactor trip and SI actuation until 30 minutes after the accident. A maximum SI flowrate is conservatively assumed for the design basis SGTR analysis in order to maximize the break flow.

The results of the thermal and hydraulic evaluation of the SGTR for the increased MSSV tolerance indicate that the primary to secondary break flow of 137,811 pounds would increase by 0.9% to 139,100 pounds. The amount of steam released for the reduction in auxiliary feedwater flow of 64,488 pounds is increased by 1.6% to 65,500 pounds. These results were used in calculating the effect of the increased MSSV tolerance on the offsite radiological consequences (see below).

Radiological Consequences

An analysis for the SGTR and evaluations for other radiological events were performed based on a ±3% tolerance on the MSSVs. The FSAR analyses for Loss of Load/Turbine Trip, Loss of Offsite Power, Locked Rotor, Main Steamline Break, and Rod Ejection remain bounding for this proposed configuration. Only the SGTR required reanalysis.

The Steam Generator Tube Rupture offsite doses were reanalyzed for the change in faulted steam generator steam releases and break flow due to the increase in tolerance. The offsite doses were calculated to be:

	Thyroid Dose	Whole Body Dose	Skin Dose
SB	3.8 rem	0.14 rem	0.17 rem
LPZ	1.6 rem	0.05 rem	0.06 rem

The doses remain within the NRC dose acceptance criteria of 10CFR100.

CONCLUSION

The evaluation shows that allowing a setpoint tolerance of ±3% does not result in any ASME code related problems with the MSSVs. Resetting the setpoint to ±1% reduces the concerns for valve leakage, operating in an unanalyzed condition, and set-pressure overlap. In addition, the steam pressures resulting from a ±3% setpoint tolerance are enveloped by existing design transients. All FSAR accident analyses continue to meet their respective acceptance criteria.

REFERENCE

a. Full Flow Certification Testing of 15 Dresser Main Steam Safety Valves for Farley Nuclear Plant, Unit II, Wyle Test Report J/N 42539.