ATTACHMENT III

SIGNIFICANT HAZARDS EVALUATION

Joseph M. Farley Nuclear Plant
Steam Generator Water Level Setpoint Changes
Associated With Lower Level Tap Relocation

Pursuant to 10 CFR 50.92 each application for amendment to an operating license must be reviewed to determine if the proposed change involves a significant hazards consideration. The amendment, as defined below, describing the Technical Specifications changes associated with modifying the steam generator (SG) water level reactor trip and safeguards actuation setpoints has been reviewed and deemed not to involve significant hazards consideration. The basis for this determination follows.

BACKGROUND

Farley Nuclear Plant currently has reactor trip and safeguards actuation on steam generator low-low level for protection from loss of heat sink caused by postulated events such as loss of normal feedwater, feedline rupture, and loss of all AC power to station auxiliaries. The current steam generator low-low water level setpoint is being revised to afford additional operating margin. The setpoint change reflects the proposed Farley modification to the lower level taps, including Farley specific instrumentation, procedures, calibration practices and uncertainties, and accounts for the increased span due to lowering of the SG lower level taps. In addition, the steam generator high-high level setpoint for turbine trip and feedwater isolation has been revised to be consistent with the increased narrow range (NR) span. WCAP-13992, "Steam Generator Lower Level Tap Relocation Assessment For J. M. Farley Nuclear Plant Units 1 and 2," provides the basis of the revised setpoints. The revision to these setpoints will allow increased operational flexibility and will reduce spurious reactor trips due to feedwater system transients.

PROPOSED CHANGE

The proposed amendment involves the change of the steam generator low-low water level reactor trip and safeguard actuation setpoint to 25% of narrow range span with a corresponding change in allowable value to 23.3%. The steam generator high-high water level setpoint change for turbine trip and feedwater isolation is 79.2% of the new narrow range span with a corresponding allowable value of 80.5%.

The lowering of the SG narrow range lower level tap and subsequent increase in the narrow range span require a proposed change in the setpoints associated with the steam generator water level protection system functions. In addition, the SG water level corresponding to the revised low-low level setpoint reflects a physical level reduction to increase operating margin between the normal operating SG level and the low-low level reactor trip setpoint.

The revised setpoints were calculated using the Westinghouse statistical setpoint methodology. The calculations included Farley-specific uncertainty allowances for process measurement accuracy and harsh environment effects. The calculations also reflected any changes to the associated safety analysis limits.

ANALYSIS

To accommodate the level tap relocation modification and the reduced steam generator inventory associated with the low-low level setpoint, several aspects of the Farley design basis required review and/or revision. All non-LOCA analyses that credit low-low steam generator level as primary protection were re-analyzed. In addition, the most limiting steamline breaks for environmental qualification which credit low-low steam generator level as primary protection were re-analyzed.

MECHANICAL EVALUATION

Level taps on the steam generator shell are connected to the steam generator without welding to the steam generator shell. The new level taps are to be installed by fastening a modified flange onto the outside surface of the steam generator using studs, nuts and spherical washers in order to provide leak tight operation in line with current recommendations for all steam generator closures. The cover assembly is fabricated from P3 type carbon steel material and consists of a transition cone welded to a circular plate. A weld connects the instrumentation piping to the flange.

The loads used for the analysis of the tap and the surrounding portion of the steam generator are the same loads used for the analysis of the original taps. The location of the taps, per the unit specific Field Change Notice (FCN), assures that the minimum distance between penetrations, specified by ASME Code, Section III, Subsection NB-3000, is maintained. Additionally, a finite element analysis of the shell area of the level tap location and the assembled level tap structure up to the pipe connection, and a bolting analysis have been performed. Loading inputs include fastener (stud/nut or bolt) preload, pressure, piping loads, and thermal transients. In all cases, the stress levels are less than the ASME Code allowables. Thus, the structural integrity of the steam generator is not adversely affected by the addition of the new level taps. The installation of the relocated level taps will not result in a fatigue usage value in the steam generator shell greater than allowable limits for the expected operating life of the steam generator. The design and installation of the level tap using the criteria of the ASME code with inherent safety factors assures that the margin of safety in the structural integrity of the steam generator shell is not reduced.

All fabrication and examination is performed in accordance with the ASME Code, Section III, Sub-Section NB (applicable editions and addenda) for NB 4332, NB 4334.1, NB 4334.2, and NB 4335. The examination acceptance criteria are defined in NB-5300. All materials meet the requirements of the ASME Code, Section III, Subsection NB (applicable editions and addenda) for NB-2331 (d) and NB-2332 (a) (2).

The piping system connecting the water level taps to the instrumentation has essentially static fluid conditions with no significant flow into or out of the steam generator. The relocation of the level taps will not have any adverse effect on the operation, function, or internal flow patterns of the steam generator. In addition, insulation design requirements for the steam generator narrow range level transmitters sensing lines will be retained by Farley to insure that there are no adverse effects on the reference leg due to high energy line breaks. The setpoint uncertainty calculations include process measurement accuracy allocations based on the relocated tap and sensing line insulation design configuration.

NON-LOCA EVALUATION

The steam generator level tap relocation and the low-low level setpoint reduction required that several Final Safety Analysis Report (FSAR) Chapter 15 accident analyses be re-analyzed. The loss of normal feedwater, loss of non-emergency AC power to plant auxiliaries, and the feed system pipe break were re-analyzed using the new setpoints associated with the level tap relocation and all acceptance criteria continue to be met.

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

Introduction

The loss of normal feedwater and the loss of non-emergency AC power to the plant auxiliaries are ANS Condition II events that are analyzed to demonstrate adequate heat removal capability exists to remove core decay heat and stored energy following reactor trip. For these events, analyses acceptance criteria include demonstrating there is no overpressurization of the primary or secondary side and that pressurizer filling does not occur. A reduction in the physical steam generator low-low level setpoint minimizes the amount of mass available following reactor trip to remove the core decay heat and stored energy, resulting in a potentially more limiting transient.

Method of Analysis

The analysis method and analysis assumptions are in concert with the current Farley FSAR. Some of the important analysis assumptions are described as follows: (1) an auxiliary feedwater flow rate of 350 gpm from two motor driven pumps at 120°F is delivered to two steam generators following an actuation signal on steam generator low-low level; (2) to maximize the pressurizer filling, the pressurizer power-operated relief valves and pressurizer spray are assumed to function; (3) the steam generator low-low water level setpoint is conservatively assumed to be at 10% NR span; (4) the initial reactor power is assumed to be at the nominal NSSS rating (2667 MWt) plus 2%; and (5) thermal design flow of 86,000 gpm/loop, supporting a 20% steam generator tube plugging level, is also assumed.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS following a reactor trip. The criterion that the pressurizer does not fill is met, assuring that the integrity of the primary system is not adversely affected. For the case without offsite power available, the results verify that the natural circulation capacity of the RCS provides sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown.

Feedwater System Pipe Break (FSAR Section 15.4.2.2)

Introduction

The feedwater system pipe break is an ANS Condition IV event which is analyzed to demonstrate that the peak primary and secondary side pressures do not exceed allowable limits and that the core remains adequately covered with water. A reduction in the steam generator low-low level setpoint minimizes the amount of mass available following reactor trip to remove the core decay heat and stored energy, resulting in a potentially more limiting transient.

Method of Analysis

Two cases are analyzed in the Farley FSAR which vary the auxiliary feedwater (AFW) delivered to the intact steam generators following actuation on a steam generator low-low level signal. The first (Case A) assumes a total AFW flow rate of 350 gpm from two motor driven pumps delivered to two steam generators 10 minutes following an actuation signal on low-low level. The second (Case B) assumes a total 150 gpm is fed to the intact steam generators on a low-low level signal following a 60 second delay. The flow is then increased to 350 gpm 30 minutes from the time of the actuation signal. Additional key assumptions are: (1) the initial power is assumed to be at 102% of the NSSS design power rating (2790 MWt); (2) a conservative core residual heat generation model based on the 1979 version of ANS-5.1 is used; (3) the steam generator low-low water level setpoint is conservatively assumed to be at 0% of the new narrow range span; and (4) a 20% steam generator tube plugging level is also assumed. The method of analysis and assumptions used are otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization of the RCS and the main steam system and to prevent core uncovery. The reactor coolant remains subcooled, assuring that the core remains adequately covered with water. The analysis results also verify that the natural circulation capacity of the RCS provides sufficient heat removal capability following reactor coolant pump coastdown.

ENVIRONMENTAL QUALIFICATION FOR SUPERHEAT OUTSIDE OF CONTAINMENT DUE TO POSTULATED MAIN STEAMLINE BREAKS

Mass and energy releases for main steamline breaks outside of containment used to calculate the environmental qualification (EQ) envelope were generated for those limiting cases that use low-low steam generator water level for primary protection for reactor trip and ESF actuation. These Farley-specific mass and energy releases were then used to determine the temperature response for environment qualification. The resulting temperature response requires no changes to the existing EQ envelope and, therefore, supports the proposed setpoint modification. All acceptance criteria continue to be met.

ADDITIONAL EVALUATIONS

All LOCA, LOCA forces, steam generator tube rupture, and LOCA related analyses have been reviewed and are unaffected by this proposed modification.

All non-LOCA analysis, including main steamline break mass and energy releases for containment response, which were not specifically re-analyzed, have been evaluated and found not to be impacted by this proposed modification.

The steam generator water level control system uses inputs from narrow range level instruments. Therefore, the control system programmed setpoint will be revised to account for the increased fluid velocity effect and the increased span resulting from the relocated lower level tap.

10 CFR 50.92 EVALUATION CONCLUSIONS

Based on the preceding evaluation, the following conclusions with respect to 10 CFR 50.92 can be reached.

1) The proposed changes to the steam generator low-low water level reactor trip and ESF actuation setpoint and to the steam generator high-high water level turbine trip and feedwater isolation setpoint do not significantly increase the probability or consequences of an accident previously evaluated

in the FSAR. Several analyses previously performed in the FSAR required re-analysis. All acceptance criteria for the re-analyzed accidents continue to be met. Therefore, there is no increase in the consequences of any previously evaluated accident. The change to the steam generator low-low water level setpoint affords additional margin to spurious trips. No fission product barriers are affected. The relocation of the steam generator lower level tap does not result in increased failure probability of the SG level tap, sensing line, or instrument. Therefore, the proposed changes to the Technical Specifications do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR.

2) The proposed changes to the Technical Specifications do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new limiting single failures o: accident scenarios have been created or identified due to the proposed changes. All safety-related systems will continue to perform as designed. No new challenges to any installed safety systems have been created by the proposed setpoint modifications. Therefore, the possibility of a new or different accident is not created.

3) The proposed steam generator water level setpoint changes do not involve a significant reduction in the margin of safety. Some re-analysis was necessary because of the proposed setpoint changes; however, all margin associated with the current acceptance criteria continues to be unaffected. The proposed design and installation of the new level taps using the criteria of the ASME Code with inherent safety factors assure that the margin of safety in the structural integrity of the steam generator shell is not reduced. Setpoint uncertainty calculations have confirmed adequate margin exists between the assumed analysis setpoints and the revised setpoints. Therefore, there is no significant reduction in the margin of safety due to the setpoint changes or the physical modification.

CONCLUSION

The changes to the FNP Technical Specifications with respect to the protection system steam generator water level setpoints do not involve a significant hazards consideration as defined by 10 CFR 50.92.

WCAP-13993

STEAM GENERATOR LOWER LEVEL TAP RELOCATION ASSESSMENT FOR J. M. FARLEY NUCLEAR PLANT UNITS 1 and 2

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FOREWORD

The management and staff at Farley have made significant progress in reducing the frequency of reactor trips by the elimination of root causes, equipment upgrades, and improved operating practices. Based on a review of the Farley trip history, it is anticipated that the steam generator narrow range level tap relocation (LTR) and a corresponding reduction of the steam generator low-low level reactor trip setpoint, which are described herein should further reduce the challenges to the protection system. The LTR and setpoint reduction will provide significant additional operating margin to better accommodate system/equipment upset conditions in the feedwater and condensate systems and to minimize unnecessary reactor trips and safeguard actuations.

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1.0 INTRODUCTION

The Farley Units 1 and 2 steam generator Level Tap Relocation (LTR) program increases the span of the narrow range protection system instrumentation by lowering the low pressure tap associated with the level transmitter variable leg sensing line. In addition, the steam generator level corresponding to the protection system low-low level setpoint will be reduced. The modification and reduced setpoint provide increased operating margin between the steam generator normal operating level and the low-low level trip setpoint. A 31 inch increase in the narrow range level operating margin is expected with the Farley Model 51 steam generators. The increased operating margin will reduce challenges to the safety systems by producing more forgiving steam generator water level control responses and will provide additional transient capabilities (loss of a feedpump without reactor trip, for example) to further improve plant reliability and availability. Also, the increased narrow range span allows quicker post-trip level recovery to help minimize cooldown effects and enhance recovery from events such as a tube rupture event by reducing isolation time of the faulted steam generator.

1.1 BACKGROUND

At Farley, the narrow range level lower tap will be moved down approximately 68 inches into the downcomer (transition cone) region of the steam generator (Figure 1-1). The steam generator level tap relocation will increase the level span from 144 inches to 212 inches. The new tap location will enhance the steam generator water level control by introducing a level indication stabilizing effect caused by the fluid velocity (velocity head) at the new tap location. In addition, the reactor trip and auxiliary feedwater start setpoint on low-low level will be reduced without changing the steam generator normal operating level. The expanded steam generator level indication span and reduced low-low level setpoint benefits are summarized below.

1. Reducing the low-low level setpoint generates a net gain in operating space between the high-high and low-low level trip setpoints which will improve plant maneuvering capabilities during startup. It is expected that the operating space will increase from the current approximately 84 inches to about 115 inches (over 36% increase) not withstanding larger instrument uncertainties and velocity head effects.

2. Relocating the lower tap to the transition cone will virtually eliminate low power steam generator shrink and swell effects. Presently when feedwater flow is increased in response to a low steam generator level, the initial instrumentation response will show an additional decrease in water level. This non-minimum phase response frequently results in over-compensation by the operating staff and the subsequent reduction of operating margin. Introducing a velocity component into the steam generator level measurement effectively masks the shrink and swell phenomena. The narrow range level response with relocated lower tap will behave more like the current wide range instrument response at low power and therefore will allow smoother manual and automatic control during startup activities.

3. The increased narrow range level span will result in relatively benign indicated water level transients during plant maneuvers and upset conditions. For example, a 50% load rejection transient typically results in a peak-to-peak swing of 43 to 45 inches in steam generator water level, which is currently equivalent to a 30% span swing with the Farley Model 51 steam generators, but with the increased span only a 15% swing would occur during this Condition I Design Basis Transient. As another example, a ±2% steady state water level oscillation should be reduced to about ±1.3%.

4. With respect to the steam generator nominal operating level, the reduction in the low-low level setpoint from 467 inches to a measured level of 428 inches (which includes allowance for instrument uncertainties and velocity head effects) results in a substantial enhancement in operating margin.

5. The increased narrow range span will also significantly reduce the time required to: 1) recover indicated water level following a reactor trip to help reduce the potential for overcooling events due to excessive use of auxiliary feedwater, and 2) isolate the faulted steam generator following a tube rupture, thus enhancing the accident recovery process and possibly reducing consequences.

The combination of increased operating space and smaller indicated transients will make feedwater control more forgiving in both the automatic and manual modes of operation.

In addition, these effects also reduce the potential for a reactor trip as a result of the loss of a main feedwater pump. It will also be more tolerant of degraded system component responses and will also provide more time for stabilizing or recovering any level transient. The increased operating margin should help reduce the number of feedwater system related reactor trips and corresponding unnecessary challenges to the Reactor Protection System. Table 1-1 provides a comparison of current parameters to those parameters expected after the level tap modification.

TABLE 1-1

FARLEY MODEL 51 STEAM GENERATOR LEVEL TAP RELOCATION

Parameters	Current	Proposed
Narrow Range Span	443 to 587 inches (Span 144 inches)	375 to 587 inches (Span 212 inches)
Nominal Level at Full Power	506 inches (44% NRS)	*498 inches (58% NRS)
Low-Low Level Setpoint	467 inches*** (17% NRS)	428 inches (25% NRS)
Hi-Hi Level Setpoint	551 inches (75% NRS)	**543 inches (79.2% NRS)
Margin to Hi-Hi Level at Full Power	45 inches	45 inches
Margin to Low-Low Level at Full Power	39 inches	70 inches
Total Operating Region	84 inches	115 inches
Improvement in Margin to Low-Low Level Trip at Full Power		31 inches (79%)
Improvement in Total Operating Region		31 inches (37%)
Nominal Level at Hot Zero Power (HZP)	490.5 inches (33% NRS)	490.5 inches (54.5% NRS)
Margin to Low-Low Level Trip at HZP	23.5 inches	62.5 inches
Improvement in Margin to Low-Low Level Trip at HZP		39 inches
Margin to Hi-Hi Level Trip at HZP	60.5 inches	52.5 inches
Reduction in Margin to Hi-Hi Level at HZP		8 inches

Note: Elevations from the top of the tubesheet.

* Actual level at 100% is 506 inches, 498 inches is measured level.

** Actual level at 100% is 552.5 inches, 543 inches is measured level.

***FNP T/S Amendment 104 (U-1) and 97 (U-2) will change this setpoint to 15%.

2.0 MECHANICAL MODIFICATION AND EVALUATION

This evaluation assesses the impact of the installation of three additional level taps on each of the Farley Units 1 and 2 Model 51 steam generators. The proposed axial locations are 20 inches above the weld between the conical shell and the lower cylindrical shell which is 375 inches above the top of the steam generator tube sheet.

The three relocated lower narrow range level taps on two steam generators in Unit 1 are below and inline with existing liquid level taps. One steam generator in Unit 2 has the liquid level taps below and inline with existing taps. The third tap on one steam generator in Unit 1 and the third tap on two steam generators in Unit 2 are moved from alignment with existing level taps to position the taps away from a shell cone seam weld or, in one case, to position the tap away from a handhole and away from the feedwater nozzle. The relocated level tap holes are approximately the same diameter as the original holes, i.e., 0.742 inches.

The ASME Boiler and Pressure Vessel Code provides criteria and requirements for evaluation of the stress levels in the primary and secondary pressure boundary for design, normal operation, and postulated accident conditions. Any modifications, repair or replacement of these components must also meet the applicable requirements of the Code so that the basis on which the unit was originally evaluated remains unchanged. Steam generator level taps represent a portion of the pressure boundary of the secondary system, and therefore, the ASME Boiler and Pressure Vessel Code criteria and requirements must be used to evaluate the stress levels in the steam generator shell for normal operating and postulated accident conditions. Section III, Subsection NB-3000 of the ASME Code provides requirements for the minimum distance between penetrations to assure the structural integrity of the steam generator shell. The margin of safety provided by use of the design pressure as a basis for pressure limits is provided by the inherent safety factors in the criteria and requirements of the ASME Code.

Mechanical Evaluation

Level taps are used on the steam generator to provide a means to connect the water level instrumentation to the steam generator without welding to the steam generator shell. The proposed level taps are to be installed by fastening a modified flange onto the outside surface of the steam generator using studs, nuts and spherical washers in order to provide leak tight operation in line with current recommendations for all steam generator closures. The cover assembly is fabricated from F3 type carbon steel material and consists of a transition cone welded to a circular plate. A weld connects the instrumentation piping to the flange.

The loads used for the analysis of the tap and the surrounding portion of the steam generator are the same loads used for the analysis of the original taps. The location of the taps, per the unit specific Field Change Notice (FCN), assures that the minimum distance between penetrations, specified by ASME Code, Section III, Subsection NB-3000, is maintained. Additionally, a finite element analysis of the shell area of the level tap location and the assembled level tap structure up to the pipe connection, and a bolting analysis have been performed. Loading inputs include fastener (stud/nut or bolt) preload, pressure, piping loads, and thermal transients. In all cases, the stress levels are less than the ASME Code allowables. Thus, the structural integrity of the steam generator is not adversely affected by the addition of the new level taps. The installation of the new level taps does not result in a fatigue usage value in the steam generator shell greater than allowable limits for the expected operating life of the steam generator. The design and installation of the level tap, using the criteria of the ASME Code with inherent safety factors, assures that the margin of safety in the structural integrity of the steam generator shell is not reduced.

All fabrication and examination are performed in accordance with the ASME Code, Section III, Subsection NB (applicable editions and addenda) for NB 4332, NB 4334.1, NB 4334.2 and NB 4335. The examination acceptance criteria are defined in NB-5300. All materials meet the requirements of the ASME Code, Section III, Subsection NB (applicable editions and addenda) for NB-2331 (d) and NB-2332 (a) (2).

The piping system connecting the water level taps to the instrumentation has essentially static fluid conditions with no significant flow into or out of the steam generator. The relocation of the level taps will not have any adverse affect on the operation, function, or internal flow patterns of the steam generator.

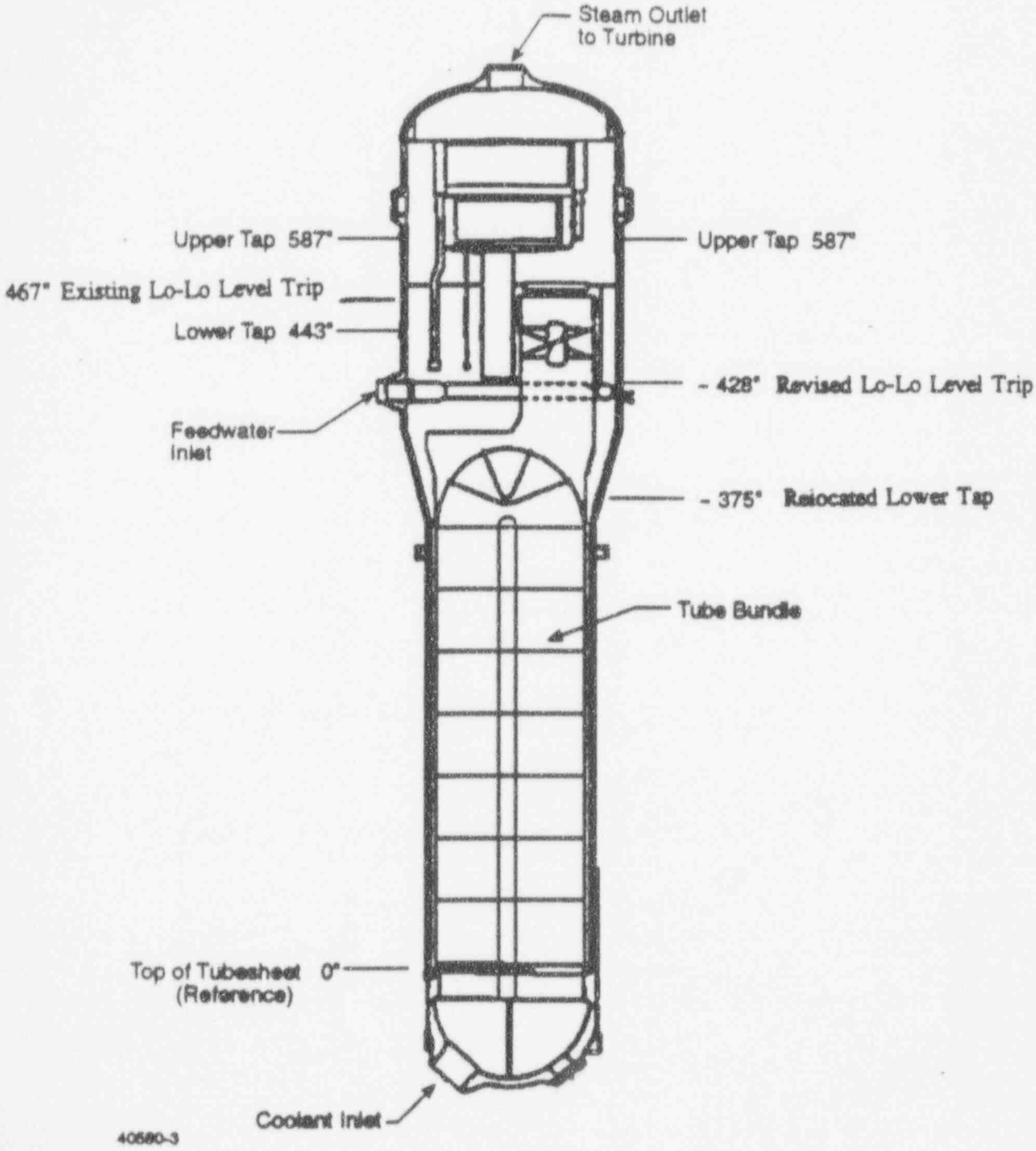

Figure 2-1 Proposed Modification

3.0 SETPOINT EVALUATION AND REFERENCE LEG INSULATION

3.1 SETPOINT EVALUATION

As part of the steam generator level tap relocation effort for Farley, setpoint uncertainty calculations were performed for steam generator level low-low level reactor trip and ESF actuation and high-high level turbine trip and feedwater isolation. The instrument uncertainties associated with each of these protection system functions were calculated using the Westinghouse statistical setpoint methodology. The calculations accounted for all known instrument uncertainty terms associated with the respanned level transmitters, signal processing equipment and calibration methods that are applicable to these functions. In addition, process measurement accuracy allowances are included to account for process pressure changes and reference leg ambient temperature changes from the reference conditions, as well as fluid velocity effects and downcomer subcooling effects associated with the new lower tap location. Furthermore, environmental allowances are included in the low-low level setpoint calculations to account for the potential effects induced on the level transmitter, signal cable and reference leg by adverse containment environmental conditions. The new Nominal Trip Setpoints of 25% narrow range span (NRS) for low-low and 79.2% NRS for high-high level trip provide positive margin to the Safety Analysis Limits, after accounting for all known uncertainties. The Allowable Values for the low-low and high-high steam generation level protection functions have been calculated to be 23.3% NRS and 80.5% NRS, respectively. These new setpoint uncertainty calculations provide the bases for the proposed technical specification changes in Section 4.7.

3.2 REFERENCE LEG INSULATION

To minimize the initial impact of high energy line breaks on the steam generator level instrumentation, i.e., delayed protection signals induced by reference leg heatup effects, Farley wrapped each narrow range sensing line with a qualified insulating material. This design feature will be retained in the level tap relocation modification, in that the re-routed variable leg sensing line will be covered with equivalent insulating materials, which are wrapped to prevent condensation damage, protected from high energy line break jet impingement forces, and qualified for post-accident containment environment. In addition, the steam generator low-low level setpoint uncertainty calculations include a Farley-specific allowance for high energy line break effects on the steam generator narrow range reference leg based on this design feature.

TABLE 3-1

STEAM GENERATOR WATER LEVEL - LOW-LOW
FEEDLINE BREAK

Parameter	Allowance*
Process Measurement Accuracy a,c	a,c
Primary Element Accuracy	
Sensor Calibration	
Measurement & Test Equipment Accuracy	
Sensor Pressure Effects	
Sensor Temperature Effects	
Sensor Drift	
Environmental Allowance Transmitter Reference Leg Heatup IR Degradation	
Rack Calibration	
Measurement & Test Equipment Accuracy	
Comparator	
Rack Temperature Effects	
Rack Drift	

*In % span (100 percent span)

Channel Statistical Allowance =

a,c

TABLE 3-2

STEAM GENERATOR WATER LEVEL - LOW-LOW
M/E RELEASE OUTSIDE CONTAINMENT

Parameter	Allowance*
Process Measurement Accuracy	
[ ] a,c	[ ] a,c
Primary Element Accuracy	
Sensor Calibration	
Measurement & Test Equipment Accuracy	
Sensor Pressure Effects	
Sensor Drift	
Environmental Allowance	
Rack Calibration	
Measurement & Test Equipment Accuracy	
Comparator	
Rack Temperature Effects	
Rack Drift	

*In % span (100 percent span)

Channel Statistical Allowance =

[ ] a,c

TABLE 3-3

STEAM GENERATOR WATER LEVEL - HIGH-HIGH

Parameter	Allowance
Process Measurement Accuracy	a,c
a,c	
Primary Element Accuracy	
Sensor Calibration	
Measurement & Test Equipment Accuracy	
Sensor Pressure Effects	
Sensor Temperature Effects	
Sensor Drift	
Environmental Allowance	
Rack Calibration	
Measurement & Test Equipment Accuracy	
Comparator	
Rack Temperature Effects	
Rack Drift	

*In % span (100 percent span)

Channel Statistical Allowance =

a,c

4.0 SAFETY EVALUATION

4.1 NON-LOCA

Farley Units 1 and 2 are similar nuclear units which share the same accident analyses. Both units have Model 51 steam generators, each of which will be modified to increase the narrow range span (NRS). The level taps, used to measure the steam generator water level during operation, are currently located at 587 inches (upper) and 443 inches (lower) above the top of the tube sheet. These tap locations result in an NRS, or a distance between the upper and lower taps, of 144 inches. The proposed modification to the steam generators is to relocate the lower tap to 375 inches above the tube sheet thereby increasing the NRS to 212 inches. The lower tap facilitates the interface between the steam generator and the level transmitter's variable leg instrument tubing, i.e., the low side sensing line. The upper tap provides the reference leg interface for the level transmitter high side sensing line. As part of this modification program, the low-low steam generator level setpoint for reactor trip and auxiliary feedwater start has been reduced.

The following accident evaluation has been prepared to justify relocation of the lower level tap on the Farley Model 51 series steam generator. The safety analysis assumptions potentially affected by the steam generator modifications include the initial steam generator water level, the low-low steam generator water level reactor trip/safeguards actuation, and the high-high steam generator water level turbine trip/feedwater isolation function. Of these, only a change to the low-low steam generator level setpoint will impact the safety analyses. No significant change to the actual nominal steam generator level control program is planned, and the current safety analyses which credit the high-high steam generator level function assume the maximum possible setpoint of 100%.

Table 4-1 identifies the current narrow range span and the revised configurations.

TABLE 4-1

CURRENT AND MODIFIED NARROW RANGE SPAN CONFIGURATION AND SAFETY ANALYSIS SETPOINTS FOR FARLEY UNITS 1 AND 2

Parameter	Current Configuration	Modified Configuration
Narrow Range Span	443 to 587 inches (span 144 inch)	375 to 587 inches (span 212 inch)
Nominal Level (@ 100% RTP)	506 inches (44% NRS)	506 inches[1] (61.8% NRS)
Low-Low Level Analysis Setpoint	443 inches (0% NRS)	375 inches[1][2] (0% NRS) 396 inches[1][3] (10% NRS) 403 inches[1][4] (13% NRS)
High-High Level Analysis Setpoint	587 inches (100% NRS)	587 inches[1] (100% NRS)

(1) Levels are actual.

(2) Low-low level setpoint assumed for "Major Rupture of a Main Feedwater Pipe" (FSAR Section 15.4.2.2) with environmental errors included.

(3) Low-low level setpoint assumed for Loss of Normal Feedwater (FSAR Section 15.2.8) and Loss of All Offsite Power to the Station Auxiliaries (FSAR Section 15.2.9) without environmental errors included.

(4) Low-low level setpoint assumed for M/E releases outside containment for EQ.

4.2 NON-LOCA ACCIDENT EVALUATION

The following evaluation justifies the relocation of the lower tap on the Farley Units 1 and 2 Model 51 steam generators to 375 inches above the tube sheet. The following sections provide discussions for each of the FSAR events.

4.2.1 Non-LOCA Transients Not Requiring Any Reanalysis

The following transients were not reanalyzed since either the transients are not affected by changes in the above mentioned safety analysis assumptions or any change to secondary side analysis assumptions will not adversely affect the results of the analyses.

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

For this ANS Condition II event, the analysis is performed at zero power conditions. A rapid reactivity addition results from the withdrawal of a bank of rods. Because of the fast nature of this event, the secondary side is not modeled. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.2 Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.2.2)

For this ANS Condition II event, various power levels and reactivity insertion rates for both minimum and maximum reactivity feedback are analyzed. The transients are terminated by either an overtemperature ΔT or high neutron flux reactor trip. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.3 RCCA Misalignment (FSAR Section 15.2.3)

For the events presented in this section of the FSAR, the DNBR criterion is applied. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.4 Uncontrolled Boron Dilution (FSAR Section 15.2.4)

This ANS Condition II event is analyzed to show that adequate time exists for operator action to terminate a dilution event prior to a loss of shutdown margin. Any changes to the secondary side setpoints have no impact on the determination of the time available for operator action. With respect to the DNBR criterion, the at-power cases, Modes 1 and 2, are bounded by the "Uncontrolled RCCA Bank Withdrawal at Power" (15.2.2). Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.5 Partial Loss of Forced Reactor Coolant Flow (FSAR Section 15.2.5)

For this ANS Condition II event, the transient is terminated by a low RCS loop flow reactor trip. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.6 Startup of an Inactive Reactor Coolant Loop (FSAR Section 15.2.6)

For this ANS Condition II event, the DNBR criterion is applied. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

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

For this ANS Condition II event, cases are analyzed at beginning and end of life conditions both with and without pressurizer control. Each of the four analyzed cases trips on either the overtemperature ΔT or high pressurizer pressure reactor trip function. Since the steam generator low-low water level reactor trip function is not actuated in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

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

For this ANS Condition II event, cases are analyzed for both full power and zero power conditions. Note that in the full power analysis, turbine trip and feedwater isolation are assumed on a high-high steam generator water level signal. The current safety analysis assumes a conservative high level setpoint for this function of 100% NRS (587 inches), which corresponds to the upper tap. This assumption has not changed for the low level tap modification. Since the steam generator low-low water level reactor trip function is not modeled in this event, the analysis is not impacted by the steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.9 Excessive Load Increase Incident (FSAR Section 15.2.11)

For this ANS Condition II event, cases are analyzed at beginning and end of life conditions both with and without automatic rod control. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.10 Accidental Depressurization of the RCS (FSAR Section 15.2.12)

For this ANS Condition II event, the transient is terminated by an overtemperature ΔT reactor trip. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impact by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.11 Accidental Depressurization of the Main Steam System (FSAR Section 15.2.13)

This ANS Condition II event is bounded by "Steam System Piping Failure" (15.4.2.1). The DNB design basis is met, and the conclusions of the FSAR remain valid.

4.2.1.12 Inadvertent Operation of ECCS During Power Operation (FSAR Section 15.2.14)

For this ANS Condition II event, the transient is initiated by a spurious safety injection signal. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.13 Complete Loss of Forced Reactor Coolant Flow (FSAR Section 15.3.4)

For this ANS Condition III event, the ANS Condition II criterion of meeting the DNBR limit is applied. The transient is terminated by an undervoltage or underfrequency reactor trip. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.14 Rupture of Main Steam Line (FSAR Section 15.4.2.1)

For this ANS Condition IV event, the ANS Condition II criterion of meeting the DNBR limit is applied. The analyses are performed at zero power with the results being primarily dependent upon secondary side parameters such as the break size, the initial steam generator inventory, steam pressure, auxiliary feedwater flow, and feedwater temperature. The actual SG level at hot zero power will not be changed by this modification; therefore, the zero power steam generator inventory assumption is not impacted. Also, since the steam generator low-low water level reactor trip function is not modeled, nor challenged, in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required and the conclusions of the FSAR remain valid.

4.2.1.15 Single Reactor Coolant Pump Locked Rotor (FSAR Section 15.4.4)

For this ANS Condition IV event, the criteria include demonstrating that peak design pressures are not exceeded and that the cladding at the "hot spot" in the core remains intact. This event is also analyzed to determine the percentage of fuel rods that experience DNB. Since the steam generator low-low water level reactor trip function is not modeled nor challenged in this event, the analysis is not impacted by the proposed steam generator modification. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

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

This is a Condition IV event and it is analyzed to show that the average fuel-pellet enthalpy at the "hot spot" remains below 200 cal/g and that the fuel melt is less than 10%. A rapid reactivity addition results from the ejection of an RCCA. Because of the fast nature of this event, the secondary side is not modeled. Therefore, no reanalysis is required, and the conclusions of the FSAR remain valid.

4.2.1.17 Main Steam Line Ruptures Inside Containment (FSAR Section 6.2.1.3.11)

The steamline break mass/energy releases inside containment are generated to ensure that the peak containment pressure and temperature limits are not exceeded. The mass/energy release data is primarily dependent upon secondary side parameters such as break size, initial steam generator inventory, steam pressure, auxiliary feedwater flow, and feedwater temperature. An increase or decrease in the steam generator inventory will result in a respective change in the mass/energy releases. The mass/energy release data presented in the FSAR is based on the current steam generator level program. Since there is no significant change proposed in the actual steam generator level (i.e., level program), no reanalysis is required, and the current mass/energy release data presented in the FSAR remains bounding.

4.3 NON-LOCA TRANSIENTS REQUIRING REANALYSIS

The following transients were reanalyzed since they are affected by changes in the steam generator low-low water level reactor trip and water level.

4.3.1 Loss of Normal Feedwater/Loss of Non-Emergency AC Power to the Plant Auxiliaries (FSAR Sections 15.2.8/15.2.9)

Introduction

These ANS Condition II events are analyzed to demonstrate that adequate heat removal capability exists to remove core decay heat and stored energy following a reactor trip. This objective is ensured by showing that there is no overpressurization of the primary or secondary side and that pressurizer filling does not occur. A reduction in the steam generator low-low level setpoint assumed in the safety analyses, to correspond to the lower level tap relocation, minimizes the amount of mass available following a reactor trip to remove the core decay heat and stored energy, resulting in a more limiting transient.

Method of Analysis

The method of analysis and analysis assumptions are in concert with the existing FSAR analyses, with important assumptions noted below. An auxiliary feedwater flow of 350 gpm at 120°F from two motor driven pumps is delivered to two steam generators following an actuation signal on low-low steam generator level is assumed. To maximize the pressurizer filling the pressurizer power-operated relief valves and pressurizer spray are assumed to function. The steam generator low-low water level setpoint is conservatively assumed to be at 10% NRS. The initial reactor power is assumed to be at the nominal NSSS rating (2667 MWt) plus 2%, and a thermal design flow of 86000 gpm/loop, supporting a 20% steam generator tube plugging level, is assumed.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS following a reactor trip. The criterion that the pressurizer does not fill is met, assuring that the integrity of the core is not adversely affected. For the case without offsite power available, the results verify that the natural circulation capacity of the RCS provides sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown.

4.3.2 Feedwater System Pipe Break (FSAR Section 15.2.8)

Introduction

This ANS Condition IV event is analyzed to demonstrate that the peak primary and secondary side pressures do not exceed allowable limits and that the core remains covered with water. A reduction in the assumed safety analysis setpoint for steam generator low-low level, due to the level tap relocation, minimizes the amount of mass available following reactor trip to remove the core decay heat and stored energy, resulting in a more limiting transient.

Method of Analysis

Two cases are presented in the Farley FSAR which vary the auxiliary feedwater (AFW) delivered to the intact steam generators following actuation on a low-low steam generator level signal. The first (Case A) assumes a total AFW flow rate of 350 gpm from two motor driven pumps delivered to two steam generators 10 minutes following an actuation signal on low-low steam generator level. The second (Case B) assumes a total 150 gpm is fed to the intact steam generators on a low-low level signal following a 60 second delay. The flow is then increased to 350 gpm 30 minutes from the time of the actuation signal. The initial power level is assumed to be at 102% of the NSSS design power rating (2790 MWt). A conservative core residual heat generation model based on the 1979 version of ANS-5.1 is used. The steam generator low-low water level setpoint is conservatively assumed to be at 0% NRS. In addition, a 20% steam generator tube plugging level is assumed. The method of analysis and assumptions used are otherwise in accordance with those presented in the FSAR.

Results and Conclusions

The transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization of the RCS and the main steam system and to prevent core uncovery. The reactor coolant remains subcooled, assuring that the core remains sufficiently covered with water. The analysis results also verify that the natural circulation capacity of the RCS provides sufficient heat removal capability following reactor coolant pump coastdown.

4.4 NON-LOCA CONCLUSIONS

Based upon the analyses and evaluations presented, the Farley Units 1 and 2 steam generator lower narrow range tap relocation and low-low level setpoint reduction can be accomplished without violating any of the conclusions of the FSAR. This includes revised safety analysis assumptions in the loss of normal feedwater, loss of offsite power, and feedline rupture events which support a reduced low-low steam generator water level setpoint to 25% NRS of the modified narrow range span.

4.5 LOCA/SGTR EVALUATIONS

The following Loss-of-Coolant Accident (LOCA) related analyses are not adversely affected by the steam generator level tap relocation at Farley: large break LOCA, small break LOCA, hot leg switchover to preclude boron precipitation, post-LOCA long term cooling subcriticality, post-LOCA long-term cooling minimum flow, steam generator tube rupture and LOCA forces. The proposed modifications will not adversely affect the normal plant operating parameters, the safeguards systems actuations nor accident mitigation capabilities important to a LOCA, nor the assumptions used in the LOCA related analyses.

4.6 MAIN STEAM VALVE ROOM TEMPERATURE RESPONSE TO SUPER HEATED STEAM

This evaluation summarizes the analysis results of a study to determine the effects of superheated steam releases, during postulated main steamline ruptures, on outside-containment equipment environmental qualification for FNP. In this study the compartment temperature and pressure profiles in the main steam valve room (MSVR), penthouse, and pipe chase due to the blowdown of main steam lines were calculated for equipment environmental qualification, using the Westinghouse COMPACT computer program. The new analyses supersede selected analyses presented in WCAP-11652 Revision 2, "Joseph M. Farley Nuclear Station Units 1 and 2 Main Steam Valve Room Temperature Response to Superheated Steam Releases." The purpose of the new analyses is in part to incorporate into the Farley outside containment main steamline break analysis the potential impact of steam generator lower level tap relocation.

The methods and assumptions identified in WCAP-11652 for calculating the compartment temperature responses to a spectrum of Main Steamline Break (MSLB) cases in the MSVR were used for these analyses. Of the break cases analyzed in WCAP-11652, only those which would be adversely affected by the proposed changes were re-evaluated here. For these break cases, the main steamline mass and energy releases were re-calculated to include the changes associated with level tap relocation. Results of these evaluations demonstrate that sufficient conservatism was available in the WCAP-11652 mass and energy releases to offset the penalty associated with the level tap relocation. An analyses setpoint of 13% NR span was used. All analyses resulted in temperatures less than 320°F, and no individual temperature profile exceeded the combined profile provided in WCAP-11652. Therefore, the proposed modification does not impact the basis for EQ outside containment in the Farley main steam valve room.

4.7 PROPOSED TECHNICAL SPECIFICATION CHANGES

TECHNICAL SPECIFICATION TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Value
Steam Generator Water Level - Low-Low	≥25% of narrow range instrument span - each SG	≥23.3% of narrow range instrument span - each SG

TECHNICAL SPECIFICATION TABLE 3.3-4

ESF ACTUATION SYSTEM INSTRUMENT TRIP SETPOINTS

Functional Unit	Trip Setpoint	Allowable Value
TURBINE TRIP AND FEEDWATER ISOLATION		
Steam Generator Water Level - High-High	≤79.2% of narrow range instrument span - each SG	≤80.5% of narrow range instrument span - each SG
AUXILIARY FEEDWATER		
Steam Generator Water Level - Low-Low	≥25% of narrow range instrument span - each SG	≥23.3% of narrow range instrument span - each SG

4.8 STEAM GENERATOR WATER LEVEL CONTROL SETPOINT CHANGES

The steam generator water level control system uses inputs from the narrow range level instruments; therefore, the control setpoints were assessed for potential impact resulting from the level tap relocation modification. Relocating the lower tap to the steam generator transition cone introduces a more significant fluid velocity component into the steam generator level measurement. The magnitude of the velocity component is a function of the fluid flow rate passing the lower tap and fluid head above the tap. As a result, the measured level will differ from the actual level, and the resultant differential magnitude varies with steam generator mass flow rate and fluid temperature and level. This fluid velocity effect is accounted for in the protection system setpoint uncertainty calculations in the process measurement accuracy uncertainty allocation. Likewise, the control system programmed setpoint will be revised to account for the fluid velocity effect in addition to the increased narrow range span. Introduction of this velocity component in the steam generator level measurement system, however, effectively masks the shrink and swell phenomena due to feedwater flow changes during low power operation. Figure 4-1 illustrates the impact of the fluid velocity effect on the measured level signal, which is demanded by the level control system programmed band, as compared to the actual and safety analysis steam generator level. Table 4-2 integrates the comparisons of the current protection and control system setpoints/parameters to the proposed setpoints, including comparisons of the measured level to actual level for each proposed change.

FARLEY LEVEL TAP RELOCATION

Programmed Water Level (Level vs. Power)

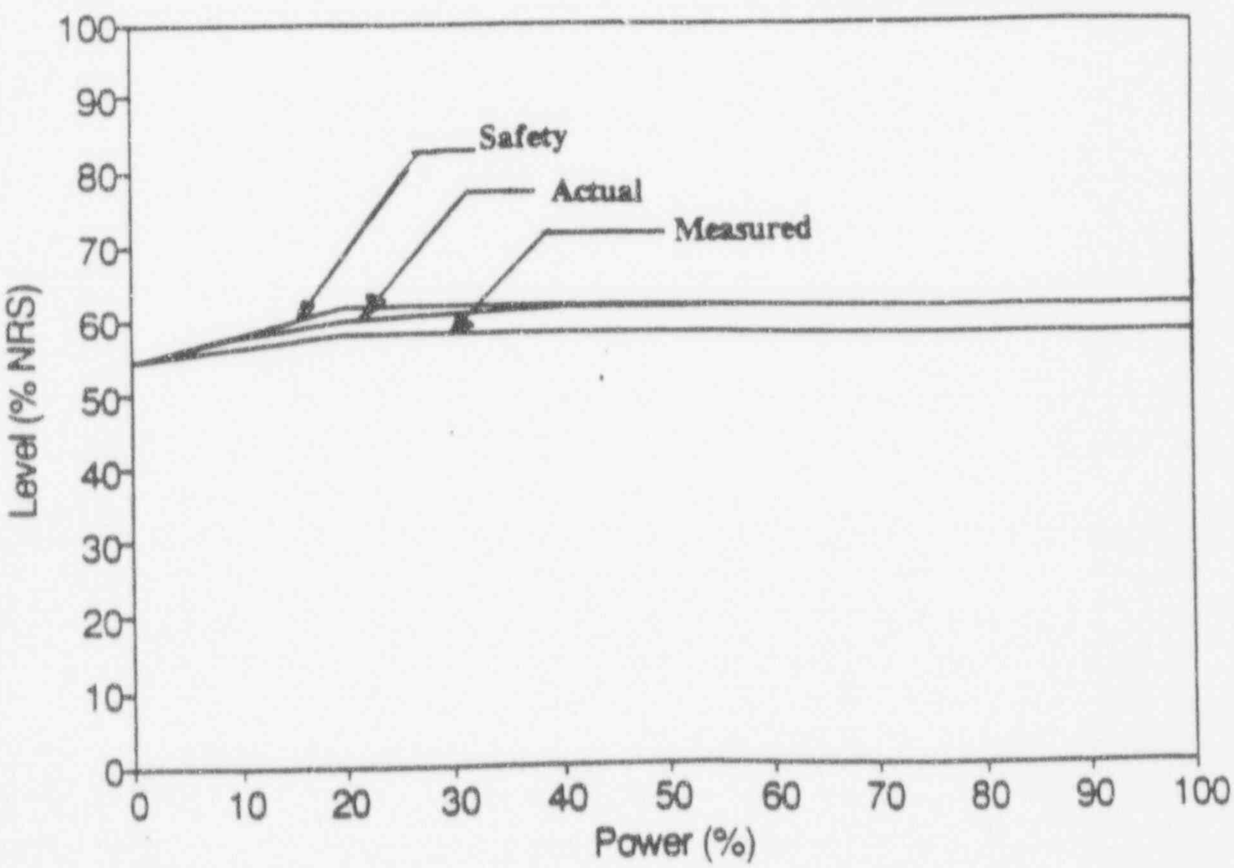

Figure 4-1 Farley LTR Level Program

TABLE 4-2
FARLEY MODEL 51 STEAM GENERATOR LEVEL TAP RELOCATION PARAMETERS

		Proposed		LTR Analysis*
Parameters	Current	Measured	Actual	Actual
Narrow Range Span	443 to 587 inch (span 144 inch)	375 to 587 inch (span 212 inch)	375 to 587 inch (span 212 inch)	375 to 587 inch (span 212 inch)
Nominal Level				
0% power	490.5 inch (33% NRS)	490.5 inch @ HZP (54.5% NRS)	490.5 inch @ HZP (54.5% NRS)	490.5 (54.5% NRS)
20% (see note 1) 100% (see note 2)	506 inch (44% NRS) 506 inch (44% NRS)	498 inch (58% NRS) 498 inch (58% NRS)	502.2 inch (60% NRS) 506 inch (61.8% NRS)	506 inch (61.8% NRS) 506 inch (61.8% NRS)
Lo-Lo Level Setpoint	467 inch (17% NRS), TS	428 inch (25% NRS); TS	432 inch at 100% Power (26.9% NRS)	
Lo-Lo Level (Safety Analysis)				375 inch (0% NRS); FLB 396 inch (10% NRS); LONF & LOOP 403 inch (13% NRS); M/E Release Outside CTMT
Hi-Hi Level Setpoint	551 inches (75% NRS); TS	543 inch (79.2% NRS); TS	552.5 inch at 100% Power (83.7% NRS)	
Hi-Hi Level (Safety Analysis)	587 inch (100% NRS)			587 inch (100% NRS)

* Safety Analysis uses actual value.
Note 1: From 0% to 20% power, level increases linearly from 54.5% to 58% NRS (for proposed modification only)
Note 2: From 20% to 100% power, level is constant at 58% NRS (for proposed modification only)
TS = Technical Specification; HZP= Hot Zero Power; FP=Full Power; NRS=Narrow Range Span

4.9 ERP SETPOINT ASSESSMENT

Farley specific Emergency Response Procedures (ERPs) setpoints have been re-calculated based on the new narrow range span resulting from the level tap relocation. The setpoint changes also considered the guidance found in both Revision 1A and Revision 1B of the Emergency Response Guidelines (ERGs).

5.0 CONCLUSIONS

The proposed level tap modification and subsequent setpoint changes have been evaluated. It is concluded that the J. M. Farley Units 1 and 2 can operate in this configuration with no unreviewed safety questions.