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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

50.59 EVALUATION SUMMARY REPORT

With this letter, the Nuclear Management Company, LLC, (NMC) submits two enclosures. Enclosure 1 contains descriptions and summaries of safety evaluations for changes, tests, and experiments made under the provisions of 10 CFR 50.59 during the period since the last update.

Enclosure 2 contains discussion of changes to regulatory commitments made within our Regulatory Commitment Change Process during the period since the last update.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

Paul W Hoffman FOR

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Site Vice President, Prairie Island Nuclear Generating Plant
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Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
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ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS, AND EXPERIMENTS – DECEMBER 2005

Below are a brief description and a summary of the safety evaluation for each of those changes, tests, and experiments which were carried out at the Prairie Island Nuclear Generating Plant by Nuclear Management Company, LLC (NMC) without prior Nuclear Regulatory Commission (NRC) approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(b).

50.59 Evaluation No. 1012 – Revise Quality Classification Process (Procedure H1) to Align with ANSI Standard 58.14 and Align QA Type to Safety Class/ASME Code Class for Piping and Related Support Systems

Description of Change

The purpose of this change is to:

- Revise quality classification process to align with American National Standards Institute (ANSI) Standard 58.14 to identify the classification process, classification criteria, and boundary interface criteria for mechanical components, and
- Align the levels of the Quality Assurance (QA) program requirements applied to safety related piping and related support systems to the Quality Classification Safety Class and related American Society of Mechanical Engineers (ASME) Code Class as follows:

Safety Class	ASME Code Class	QA Type
SC 1	1	IA
SC 2	2	IB
SC 3	3	IC

This change does not revise the requirements of QA Types IA, IB, and IC as specified within the Engineering Manual and the Welding Program Manual. It does implement a change in the application of those requirements to piping systems and subsystems that is based on compliance with industry standards for system Quality Classification and the Quality Assurance principle that the highest level of quality assurance program requirements is applied to the highest level (as defined by Safety Class) of safety related systems.

This Evaluation **SHALL NOT** be used alone as a basis to change the quality classification of any Structure, System, or Component (SSC) - that process is controlled by the requirements of Prairie Island procedures.

Summary of 50.59 Evaluation

The activities of revising the quality classification process to align with ANSI Standard 58.14 and aligning quality assurance program levels to Safety Class/ASME Code Class does not introduce the possibility of a change in the frequency of an accident, or the likelihood of occurrence of a malfunction of a System, Structure or Component (SSC) important to safety, or increase the consequences of an accident, or the increase in consequences of a malfunction of an SSC important to safety, or create the possibility for an accident of a different type, or create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the Updated Safety Analysis Report (USAR) because the activity is not an initiator of any accident and no new failure modes are introduced. The activity also does not result in a design basis limit for a fission product barrier (DBLFPB) as described in the USAR being exceeded or altered.

50.59 Evaluation No. 1016 – Dynamic Rod Worth Measurement Technique

Description of Change

This change is to add the Westinghouse Dynamic Rod Worth Measurement (DRWM) technique to the Prairie Island startup physics testing performed after refueling. DRWM is a method of measuring the reactivity worth of individual control and shutdown banks. DRWM is accomplished by inserting and withdrawing the bank at the maximum stepping speed, without changing boron concentration, and recording the signals on the excore detectors. The recorded signals are processed on the reactivity computer, which solves the inverse point kinetics equation with proper analytical compensation for spatial effects. DRWM has been reviewed and approved by the NRC for measurement of rod worth at the beginning of reload cycles for two, three, and four loop Westinghouse cores.

Summary of 50.59 Evaluation

The DRWM technique has been reviewed and approved by the NRC, provided the limitations and restrictions described in the Safety Evaluation Report (SER) are met. This 50.59 Evaluation demonstrates that the DRWM technique approval is applicable for use at Prairie Island, and that all limitations and restrictions of the SER are met.

50.59 Evaluation No. 1018 – Residual Heat Removal & Component Cooling Heat Exchanger Capability During Post – Loss of Coolant Accident (LOCA) Recirculation

Description of Change

The purpose of this 50.59 Evaluation is to review safety analyses of post-LOCA containment response. New CONTEMPT analyses were performed in support of more detailed evaluations of heat transfer during post-LOCA recirculation. Rerunning safety analyses to demonstrate that required safety functions and design requirements are met requires a 50.59 Evaluation. No other changes are being made by this evaluation. In the procedures for transfer to recirculation, the operations personnel are directed to remove the component cooling (CC) flow control valve stops. This procedural step was originally added to enhance the ability of the cooling water (CL) system to support safeguards equipment operation during post-LOCA mitigation; specifically, the Containment Fan Coil Units (CFCUs). This 50.59 Evaluation shows that, provided the valve stop is removed within the first 24 hours following the accident, acceptable results would be obtained. The action to remove the valve stop is accomplished within 24 hours. Appropriate USAR updates are included in this evaluation.

Summary of 50.59 Evaluation

The Residual Heat Removal and CC heat exchanger analysis and the associated containment integrity analyses evaluate post-accident response and are not accident initiators; thus, there is no increase in frequency of any accident or creation of an accident of a different type. Approved methods of evaluation are used for the containment integrity analyses performed to evaluate these changes. The results from these analyses show that the containment peak pressure is maintained below the design basis limit of 46 pounds per square inch gage (psig); therefore, a DBLFPB is not exceeded or altered. The predicted containment pressure profile is less than that assumed for determining the containment leakage rate in the dose analyses. Thus, there is no change to the consequences of any accident analyses or due to any equipment malfunctions. The results from the containment integrity analyses and any other affected analyses were reviewed to ensure that equipment important to safety would not be adversely affected. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

50.59 Evaluation No. 1019 – Generic Letter 96-06; Containment Fan Coil Unit Two Phase Flow

Description of Change

Sections of the USAR that discuss heat removal from containment by the CFCU are being changed to include a discussion regarding boiling at the outlet of the CFCU. This information is a summary of evaluations/calculations performed in support of responses to Generic Letter 96-06. USAR Figure 10.4-6 is being updated to reflect the results of the latest calculation. Other changes to supporting sections are also being made. The

CFCUs and associated piping have been evaluated under design basis accident conditions. Using the conservative assumptions of these bounding analyses, it is predicted that there will be boiling in the CFCUs and two-phase flow in the downstream piping. The heat removal capacity of one train of CFCUs exceeds the minimum heat removal capability used in the containment integrity analysis. Additionally, other potential effects of two-phase flow do not impact the integrity of the downstream piping. Therefore, the CFCUs and the downstream piping are able to perform their design basis functions.

Summary of 50.59 Evaluation

Two phase flow in the CFCU would be the result of an accident and, as such, has no impact on the frequency of occurrence of an accident. The likelihood of occurrence of a malfunction of an SSC important to safety is less than minimal because the CFCU heat removal rate continues to exceed the minimum value assumed in safety analyses, which ensures containment peak pressure remains below the acceptance criteria. Additionally, the effects of two phase flow erosion or corrosion would be so small that they would not be measurable or discernable. Thus, cooling water system piping integrity is not impacted. For the same reasons, two phase flow does not create the possibility for a malfunction of an SSC important to safety with a different result. Since no new detrimental effects are created, the possibility of an accident with a different result is not created. The calculations demonstrate that peak containment pressure will not exceed the acceptance criteria in safety analyses and therefore the assumptions used to predict off site dose rates do not change. Therefore, there is no change in the consequences of an accident or a malfunction of an SSC important to safety. Also, since the calculations demonstrate that peak containment pressure will not exceed the acceptance criteria in safety analyses, no DBLFPB is altered or exceeded.

50.59 Evaluation No. 1020 – Unit 1 Replacement Steam Generator (RSG) - Main Steam Line Break Containment Response

Description of Change

The purpose of this evaluation is to review the Main Steam Line Break (MSLB) containment analysis performed by Framatome ANP (FRA-ANP) for operation of Unit 1 with the Replacement Steam Generators. The analysis was performed using FRA-ANP's version of the CONTEMPT code to calculate the containment pressure and temperature response.

Summary of 50.59 Evaluation

The analysis of the containment response to a MSLB is not an accident initiator; thus, there is no increase in frequency of any accident or creation of an accident of a different type. The results from the containment response analysis show that the containment peak pressure is maintained below the design basis limit of 46 psig; therefore, a DBLFPB is not exceeded or altered. Containment integrity is maintained; thus, the dose analysis for the MSLB inside of containment is still bounded by the MSLB outside of containment and there are no changes to the consequences of any accident analyses or equipment malfunction. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result. The methods used for performing these analyses are not a departure from a method of evaluation described in the USAR.

50.59 Evaluation No. 1021 – Unit 1 RSG – Reactor Coolant Loop Structural Evaluation

Description of Change

The purpose of this evaluation is to review the Reactor Coolant Loop Structural Evaluation performed by FRA-ANP for operation of Unit 1 with the Replacement Steam Generators.

Summary of 50.59 Evaluation

There is no change to the configuration of the Reactor Coolant System (RCS) loops and the structural analysis demonstrates that the current design basis criteria and limits continue to be met, thus there is no increase in frequency of any accident or creation of an accident of a different type. The methods used for performing these analyses are not a departure from a method of evaluation described in the USAR. The results show that the structural criteria and limits are maintained, thus the safety analyses are unaffected and there is no affect to the consequences of any accident analyses or equipment malfunction.

50.59 Evaluation No. 1022, Addendum 1 – Consequence Analysis for Reactor Vessel Head and Upper Internals Drop on Open Fueled Reactor

Description of Change

Currently heavy load lifts over the open fueled reactor vessel are performed with containment closed with the intent that the LOCA safety analysis bounds the consequences of a load drop on the fuel. Analysis has been performed to demonstrate that dropping the reactor vessel head will not damage the fuel, and dropping the upper internals with the reactor containing fuel that has been subcritical at least seven days will not result in unacceptable consequences with containment open. This analysis permits movement of the reactor upper internals over the open fueled reactor vessel with containment open to the outside atmosphere under the condition that the reactor has been shutdown for at least seven days. In the event of an accident, containment closure would be required within one hour. Dropping the head will not result in fuel damage and therefore does not require containment boundary restrictions. Implementation of the change will require revision of NRC Commitment OTH007737, USAR section 12, and numerous maintenance and operating procedures regarding containment boundary control and the movement of the reactor vessel head and upper internals.

Summary of 50.59 Evaluation

The analysis performed demonstrates that the change in the administrative controls regarding containment closure results in a less than minimal increase in the consequences of an accident previously evaluated. The increase in predicted dose was less than 10% of the difference in the fuel handling accident dose results and the limits specified in Regulatory Guide (RG) 1.183, and does not exceed the Standard Review Plan guideline value for this event (as defined in RG 1.183).

The consequences of a malfunction of an SSC important to safety are not changed because the analysis includes single active failure and does not credit any other SSCs to mitigate consequences.

Changing the administrative controls regarding containment closure has no impact on the frequency or the likelihood of occurrence of an accident or a malfunction, nor does it create the possibility of an accident or a malfunction of a different type.

The analysis does not involve any design basis limits for fission product barriers. Fission product barriers are part of the analysis, but no limits are changed.

The analysis was performed using an NRC reviewed and approved methodology; therefore it is not a departure from an approved method.

50.59 Evaluation No. 1023 – Beacon Flux Map Processing Code

Description of Change

This evaluation is to allow the use of the existing power distribution measurement uncertainties of 1.05 (F_Q) and 1.04 ($F^N_{\Delta H}$) using the BEACON-INCORE flux map processing code. Currently the basis for these uncertainty values is given in Westinghouse WCAP-7803, based on the INCORE code. Compared to INCORE, BEACON is an improved method of processing flux map data. The result of this evaluation is to add the BEACON topical report, WCAP-12472, as an additional reference in Technical Specification Bases section 3.2.1

Summary of 50.59 Evaluation

The determination that it does not constitute a departure from a method of evaluation is based on the NRC approval of BEACON, and it's applicability to Prairie Island. The NRC compared the results between the two methods used. The results demonstrate that BEACON yields essentially the same results as INCORE. As such, continued use of the uncertainty factors of 1.05 (F_Q) and 1.04 ($F^N_{\Delta H}$) are justified. The NRC has accepted the similarity of the results in their approval of WCAP-12472.

50.59 Evaluation No. 1025, Revision 1 – Zebra Mussel Treatment

Description of Change

Chemically treat portions of the Circulating Water System (CW), the Cooling Water System and Fire Protection System (FP) to eradicate zebra mussel population within the Prairie Island facility. In 2004, a treatment was performed, however, since 2004, Prairie Island staff has observed an increase in zebra mussel population within the Circulating Water Bays. This evaluation is being revised to incorporate results of a new calculation and to evaluate the effects of an increased mussel population.

While simultaneous treatment of Unit 1/Unit 2 for zebra mussels creates the potential to challenge plant systems, the design of the plant screens, strainers, and support systems, and procedural controls minimize the potential for plugging. The performance of the zebra mussel treatment procedure SHALL be considered a special test/procedure and as such requires evaluation prior to performance.

Summary of 50.59 Evaluation

The treatment for zebra mussel control has the potential to affect the cooling water, circulating water and fire protection systems. The evaluation has determined that this activity does not result in more than a minimal increase in the frequency of occurrence of an accident or the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR or any pending submittal. It has also been determined that there will be no affect on off-site or on-site dose resulting in more than a minimal increase in the consequences of an accident or malfunction of an SSC important to safety previously evaluated in the USAR or any pending submittal. The evaluation shows that the activity does not create a possibility for an accident of a different type than has already been evaluated in the USAR and pending submittals and that there are no new failure modes that are not already bounded by existing analyses that would result in a possibility for a malfunction of an SSC important to safety with a different result than previously evaluated. Finally, the activity has been found to not result in a DBLFPB being exceeded or altered.

50.59 Evaluation No. 1026 – Unit 1 RSG - Stress & Fatigue Analysis Report

Description of Change

The purpose of this evaluation is to review the FRA-ANP calculations associated with the Stress and Fatigue Analysis Report for the Unit 1 Replacement Steam Generators. These calculations are provided to support the design change 00SG02.

Summary of 50.59 Evaluation

The structural analysis demonstrates that the RSGs meet the current design basis criteria and limits continue to be met, thus there is no increase in frequency of any accident or creation of an accident of a different type. The changes to methods of analyses that are used are approved by the NRC. The results show that the structural criteria and limits are maintained, thus the safety analyses are unaffected and there is no affect to the consequences of any accident analyses or equipment malfunction.

50.59 Evaluation No. 1027, Revision 1 – Design Change 00SG02 Revision 0 - Unit 1 RSGs

Description of Change

Design Change 00SG02 provides the basis for the replacement of the Unit 1 Steam Generators. The RSGs are provided by FRA-ANP. The RSGs are FRA-ANP Model 56-19 steam generators that are designed to replace the Westinghouse Model 51 steam generators. The design change provides the analysis and documentation to allow operation of Unit 1 with the RSGs.

Summary of 50.59 Evaluation

Design Change 00SG02 does not increase the frequency of occurrence of an accident. All design basis accidents were reviewed. Only Steam Generator Tube Rupture (SGTR) is directly linked to the RSGs. The analysis demonstrates that the RSGs do not increase the frequency of occurrence of a SGTR. The RSGs will not increase the likelihood of occurrence of a malfunction of equipment.

Dose consequence accidents were evaluated with the RSGs; the evaluations show that there is no more than a minimal increase in release of secondary activity due to an increase in primary volume.

Design Change 00SG02 does not create the possibility for an accident of a different type because the RSG design meets the ASME Code requirements and the only accident that is initiated by the RSGs is SGTR that is already described in the USAR. Design Change 00SG02 also does not increase the possibility of a malfunction of equipment different than previously evaluated in the USAR.

Design Change 00SG02 does not affect a DBLFPB. The analysis demonstrates that with RSGs installed, no DBLFPB is exceeded.

Design Change 00SG02 does not result in a departure from a method of evaluation.

50.59 Evaluation No. 1028 – RSG - Main Steam Line Break Mass & Energy Release for Containment Response

Description of Change

The purpose of this evaluation is to review the containment MSLB mass and energy release analysis performed by FRA-ANP for operation of Unit 1 with the RSGs. The analysis was performed using FRA-ANP's RELAP5/MOD2-B&W (RELAP5) computer code to generate the mass and energy releases from the secondary system.

Summary of 50.59 Evaluation

The inputs/methods/analysis for generating mass and energy releases during a MSLB are not accident initiators; thus, there is no increase in frequency of any accident or creation of an accident of a different type. The results from the containment response analysis (50.59 Evaluation No. 1020) show that the containment peak pressure is maintained below the design basis limit; therefore, a DBLFPB is not exceeded or altered. Containment integrity (50.59 Evaluation No. 1020) is maintained; thus, the dose analysis for the MSLB inside of containment is still bounded by the MSLB outside of containment and there are no changes to the consequences of any accident analyses or equipment malfunction. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result. The methods used by FRA-ANP are appropriate and approved for the intended application.

50.59 Evaluation No. 1030 – Unit 1 Cycle 22 Westinghouse Transition COLR Revision (Revision 1 and 2)

Description of Change

This activity is to transition from a NMC Nuclear Analysis Department (NAD) performed non-LOCA safety analysis to a Westinghouse performed analysis. This change will result in major changes to the USAR, mainly in Chapter 14. It will also require a revision to the Core Operating Limits Report (COLR), which is currently based on the NAD analysis.

In order to implement this change, an amendment to the Prairie Island operating license was required. This amendment was issued by the NRC on April 28, 2004 as License Amendment 162/153 for Units 1 and 2, respectively.

Summary of 50.59 Evaluation

This Evaluation demonstrates the acceptability of revision to the Unit 1 Cycle 22 COLR using NRC approved Westinghouse methods as described in Technical Specification (TS) 5.6.5. These methods were incorporated into TS 5.6.5 under License Amendment 162/153. As part of the NRC review of the License Amendment Request, NMC submitted the results of the analysis to demonstrate the acceptability of the methods. In the SER granting the License Amendment, the NRC concluded that the analyses were acceptable because they demonstrated that all applicable acceptance criteria were met. Using the methodology described in WCAP-9272-A, Westinghouse demonstrated that the analyses (which were reviewed and found acceptable) bounded operation on Unit 1 Cycle 22. Based on the NRC review and the Westinghouse Reload Safety Evaluation the new values for the COLR are demonstrated to be acceptable.

50.59 Evaluation No. 1031 – Unit 2 Cycle 22 Westinghouse Transition COLR Revision (Revision 1 and 2)

Description of Change

This activity is to transition from an NMC NAD performed non-LOCA safety analysis to a Westinghouse performed analysis. This change will result in major changes to the USAR, mainly in Chapter 14. It will also require a revision to the COLR, which is currently based on the NAD analysis.

In order to implement this change, an amendment to the Prairie Island operating license was required. This amendment was issued by the NRC on April 28, 2004 as License Amendment 162/153.

Summary of 50.59 Evaluation

This Evaluation demonstrates the acceptability of revision to the Unit 2 Cycle 22 COLR using NRC approved Westinghouse methods as described in Technical Specification 5.6.5. These methods were incorporated into TS 5.6.5 under License Amendment 162/153. As part of the NRC review of the License Amendment Request, Prairie Island submitted the results of the analysis to demonstrate the acceptability of the methods. In the SER granting the License Amendment, the NRC concluded that the analyses were acceptable because they demonstrated that all applicable acceptance criteria were met. Using the methodology described in WCAP-9272-A, Westinghouse demonstrated that the analyses (which were reviewed and found acceptable) bounded operation on Unit 2 Cycle 22. Based on the NRC review and the Westinghouse Reload Safety Evaluation the new values for the COLR are demonstrated to be acceptable.

50.59 Evaluation No. 1032 – Revised Containment Integrity Analysis with New M&E Methods

Description of Change

The purpose of this activity is to perform a new analysis of record for large - break LOCA (LBLOCA) containment integrity (USAR Appendix K analysis). The new containment integrity analysis will utilize the existing licensing basis containment evaluation model for LOCA as described in the USAR, Revision 26, Appendix K (CONTEMPT/LT-028). All specific elements used in the CONTEMPT evaluation model are identical to the existing USAR Rev 26 Appendix K analysis. In addition, all inputs are identical with the exception of new limiting mass and energies calculated by Westinghouse to bound both the original and new Framatome Model 56/19 Steam Generators and the mechanistic changes necessary to align CONTEMPT modeling with the mass and energy (M&E) analysis results. These new M&E calculations use a method of evaluation based on Westinghouse WCAP-10325. This is a new method of evaluation for LOCA mass and energy calculations for Prairie Island. The results of the new analysis of record using the new method of evaluation are within the existing containment licensing basis design criteria specified in the USAR, Revision 26, section

5.2.1.1 and the LOCA dose analysis documented in the USAR, Revision 26, section 14.9.

Summary of 50.59 Evaluation

The M&E releases following a LBLOCA and the resultant containment integrity analysis are used to evaluate post accident response and are not accident initiators. Therefore, there is no increase in frequency of an accident or creation of an accident of a different type. Methods approved for the intended application are used for the generation of mass and energy releases and the associated containment integrity analysis performed to evaluate these changes. The results from these analyses show that the containment peak pressure is maintained below the maximum pressure limit of 46 psig; therefore, a DBLFPB is not exceeded or altered. The containment shell and dome metal temperatures are predicted to remain below that assumed in the vessel stress analyses (268 degrees F). In addition, the predicted containment pressure profile is less than that assumed for determining the containment leakage rate in the dose analysis. Thus, there is no change to the consequences of any accident analysis due to any equipment malfunctions. Since no assumptions based on plant configuration changes were incorporated into this new analysis, exclusive of the new M&E's, other analysis would not be invalidated as a result of this change. The new M&E method of evaluation does not impose any new operational requirements or make any assumptions inconsistent with the existing plant configuration. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

50.59 Evaluation No. 1034 – RSG - Low Temperature Overpressurization (LTOP) Analysis

Description of Change

The purpose of this evaluation is to review the LTOP mitigation system performance analysis performed by FRA-ANP for operation of Unit 1 with the Replacement Steam Generators.

The analysis in the FRA-ANP calculation was performed using FRA-ANP's RELAP5 computer code to demonstrate that the LTOP system along with the current setpoints will meet the criteria listed in the USAR, i.e. maximum RCS pressure less than 110% of the 35 Effective Full Power Years (EFPY) most limiting steady state curve (Unit 1) 10 CFR Part 50 Appendix G pressure/temperature limitations or less than 800 psig at the power operated relief valve discharge piping, whichever is more limiting.

Summary of 50.59 Evaluation

The inputs/methods for the LTOP analysis are not accident initiators; thus, there is no increase in frequency of any accident or creation of an accident of a different type. The results of the analysis show that the acceptance criteria continues to be met and thus the RCS integrity is maintained and there is no impact on the consequences of any accident analyses or equipment malfunction. Therefore, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result. The methods used by FRA-ANP are appropriate and approved for the intended application.

50.59 Evaluation No. 1035 – Compensated Hi Tavg Parameter Changes

Description of Change

Occasionally the Rod Control System will automatically step rods out or in a single step based on control signals from the Reactor Control system. This is caused by the high lead/lag gain ratio in the Hi-Tavg Compensator Lead/Lag Unit Foxboro module. This gain was initially set at 8 per Westinghouse WCAP-7721 for original plant start-up with a note that the gain may be adjusted based on optimum plant operating conditions. A single step from the controlling bank of rods is not significant from a reactivity change perspective. Current expectations are that all reactivity changes be controlled. Since it is not acceptable to routinely operate Rod Control in manual, it was decided to pursue a reduction in the lead/lag gain ratio. Westinghouse was contracted to perform an analysis and make recommendations for the Hi-Tavg Compensator Lead/Lag Unit time constant changes to minimize occasional rod stepping.

In order to minimize the potential for rod stepping to occur due to the Tavg signal noise, Westinghouse recommends that the Hi-Tavg Compensator Lead/Lag Unit time constants be revised. The current values of compensation are 80/10/4/2 (from WCAP-7721) and are recommended for revision to 40/10/8/2. This will physically be implemented by changing the following parameters in the TM-401H Foxboro module in both Unit 1 and Unit 2: The lead/lag gain ratio K=8 will be changed to K=4 and lag constant T2=4 will be changed to T2=8. These recommended values have been implemented at other plants that have noted similar rod stepping problems due to RCS temperature variations. The revised time constants are expected to minimize the potential for automatic rod stepping in the presence of steady-state RCS temperature variations.

The Rod Control System is a non-safety related system designed for maintaining RCS temperature for various transients. It is not relied upon for accident mitigation. Changes to these time constants do not adversely affect any safety analysis.

Summary of 50.59 Evaluation

Since this is a change to time constants in an existing control system and is within the operating bounds of that system, this change does not involve the increase in frequency or likelihood of an accident, or consequences of a malfunction, an accident of a new type or a malfunction with different results than previously evaluated in the USAR.

These time constant changes were evaluated using Westinghouse methodology approved for Prairie Island so there is no change in the method of evaluation.

Per the Westinghouse analysis, these time constant changes will not result in exceeding a DBLPFB as described in the USAR.

Per the Westinghouse Evaluation, these time constant changes do not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR.

50.59 Evaluation No. 1036 – Unit 1 Cycle 23 Core Reload

Description of Change

This design change is required to allow for continued power operation of Prairie Island Unit 1 for approximately 18 months. The fuel in the current core will be burned to a state that no longer allows for full power operation. This reload will replace burned fuel from Unit 1 Cycle 22 with 49 fresh fuel assemblies. This will allow the Unit 1 reactor to produce power at its rated capacity. The Cycle 23 design uses fresh assemblies of the Westinghouse Vantage+ design. For peaking and isothermal temperature coefficient (ITC) control the design uses 396 gadolinia pins at 8 w/o.

The Cycle 23 analyses in support of this 50.59 Evaluation all bound operation with the Framatome Model 56/19 replacement steam generators.

Cycle 23 will also replace the existing control rods with new Enhanced Performance Rod Cluster Control Assemblies (EP-RCCAs). The replacement EP design is similar to the existing rods with the exception of a slight change in the length of the chrome plating on the outside of the absorber rods.

Cycle 23 is expected to attain an end of cycle exposure of 20,651 megawatt days per metric ton uranium (MWD/MTU) based on the expected startup and shutdown dates.

Summary of 50.59 Evaluation

The evaluations demonstrate that the Prairie Island Unit 1 Cycle 23 reload design and associated COLR do not result in the accepted safety limits for any accident being exceeded. The Cycle 23 design is consistent with the description of the core in the USAR. The core contains 121 fuel assemblies using a 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the USAR. The only change from Cycle 22 is the distribution of new and used assemblies. This results in a redistribution of the isotopic distribution of the core which changes the core physics parameters of the reactor. The effect of these changes in the cycle physics parameters on cycle operation and accident analyses have been evaluated using NRC approved methods.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore the reload modification for Unit 1 Cycle 23 is safe and consistent with Prairie Island's current licensing basis.

50.59 Evaluation No. 1037 – Affect of Revised Unit 1 Main Steam Line Break on Auxiliary Building Environment

Description of Change

The changes covered by this 50.59 Evaluation are new analyses that determine the effect of a Unit 1 MSLB, following replacement of the original Unit 1 steam generators, on the environment in the Auxiliary Building. These analyses differ from previous analyses in the four respects:

- Mass & Energy Releases Whereas previous pipe break mass and energy release calculations assumed the reactor would trip immediately following a MSLB, the new analysis conservatively assumes a trip after 10 minutes. As a consequence, more mass and energy is released into Auxiliary Building. Also, due to the replacement steam generators, mass flow (pounds per second) and energy (btus per pound) released during a MSLB are higher. As a consequence, steam generator tubes are uncovered 52 seconds and 12 percent sooner.
- Heat Sinks Previous Auxiliary Building high energy line break (HELB) analyses only took credit for 250,800 square feet of concrete heat sinks, although input documents also identified 1,800 square feet of metal grating. This analysis takes credit for 250,800 square feet of concrete heat sinks plus 175,000 square feet of metal heat sinks including 1,800 square feet of grating.
- Volumes Net Auxiliary Building compartment volumes used in the previous HELB analyses increased by 0.025 percent to 925,500 cubic feet. These adjustments resulted from additional information obtained while inventorying metal heat sinks in the Auxiliary Building.
- Initial Temperatures Compartment temperatures seconds prior to the postulated pipe break were reduced from 130 degrees F to 105 degrees F, which is the maximum temperature of the Auxiliary Building based on the cooling capacity of the Auxiliary Building Normal Ventilation System.

Summary of 50.59 Evaluation

The new HELB analysis found that, with Unit 1 replacement steam generators in service, peak Auxiliary Building accident pressures would generally be the same as before and temperatures would decrease by 10 degrees F. Where pressure increased, it would only increase from 0.01 to 0.02 psi. Safety related block walls would be most affected. Factoring in pressure increases, the reserve capacity of safety related block walls would be 20 percent or more. The exception would be one block wall with a reserve capacity of two percent. However, if more refined modeling techniques and computer analyses were used to reduce conservatism, the wall's reserve capacity would increase substantially. Only two areas of the Auxiliary Building, Compartments L1 & L2, would see a temperature increase. The increase would be 2 degrees F, and neither area houses environmentally qualified equipment.

Since pressure and temperature increases would be minimal, the likelihood of a malfunction resulting from the increases would be minimal. Likewise, changes to Auxiliary Building environment would not be capable of initiating malfunctions with results different than already stated in the USAR. Methods used in the five analyses covered by this evaluation are consistent with those identified in the USAR.

50.59 Evaluation No. 1040 – Square Root Sum of Squares (SRSS) Load Combination

Description of Change

This evaluation is limited to addressing a change in the methodology used to combine seismic and LOCA loads for faulted conditions in stress analyses of components designed to ASME Code Class 1, 2, or 3 requirements.

USAR Section 12.2.1.5, "General Design Criteria for Components", and Table 12.2-11, "Load Combinations for Components" provide Methods of Evaluation for combining loads for normal operating conditions with seismic and pipe rupture loadings. For the combination of Normal, Design Basis Earthquake (DBE) and Pipe Rupture, the USAR specifies direct summation of loads due to dead, live, DBE and Design Basis Accident (DBA) loads. An alternate method has been approved by the NRC to allow the DBE and LOCA loads to be combined using the SRSS methodology.

The proposed change activity is to modify the USAR to allow the use of either the direct summation method or the SRSS methodology.

Summary of 50.59 Evaluation

The method of evaluation change (SRSS combination of seismic and LOCA loads as described in NUREG-0484 Revision 1) has been previously approved by the NRC in NUREG-0843. The methods used are within the constraints of NUREG-0484 Revision 1, namely the design of the component is to ASME Code Class 1 and the use of a linear elastic dynamic analysis to meet the appropriate ASME Code Service Limit. Therefore, the SRSS combination of seismic and LOCA loads in accordance with NUREG-0484 Revision 1 does not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.

50.59 Evaluation No. 1042 – Reactor Vessel Missile Shield Replacement

Description of Change

This evaluation only addresses replacing the existing reactor vessel missile shield with one that is integrated into the head assembly upgrade package (HAUP).

Summary of 50.59 Evaluation

The analyses evaluated the local and global damage from a missile generated by the failure of a Control Rod Drive Mechanism housing and demonstrated that the reactor vessel missile shield will continue to perform its design function of protecting plant components and structures. Analyses also show that the missile shield and all supporting members will continue to perform their design functions following a seismic event. The heavy load program already addresses operation without the reactor vessel missile shield covering the pressurizer vault. Therefore the proposed changes do not increase the frequency of any accidents or malfunctions, nor do they create any accident/malfunction of a different type. The results of the analyses show that the structural criteria and limits are maintained, thus the safety analyses are unaffected and there is no affect on the consequences of any accident analyses or equipment malfunctions. Replacing the missile shield does not involve a design basis limit for a fission product barrier. The methods used in the analyses do not result in a departure from a method of evaluation described in the USAR.

50.59 Evaluation No. 1046, Revision 1 – Unit 2 Cycle 23 Core Reload

Description of Change

This design change is required to allow for continued power operation of Prairie Island Unit 2 for approximately 18 months. The fuel in the current core will be burned to a state that no longer allows for significant full power operation. This reload will replace burned fuel from Unit 2 Cycle 22 with 45 fresh fuel assemblies. This will allow the Unit 2 reactor to produce power at its rated capacity. The Cycle 23 design uses fresh assemblies of the Westinghouse Vantage+ design. For peaking and ITC control the design uses 460 gadolinia pins at 8 w/o.

This is Revision 1 to Evaluation No. 1046. This revision incorporates changes to the core design resulting from the decision to insert 45 fresh fuel assemblies in Cycle 23, versus the 49 fresh assembly design evaluated in Revision 0 of this Evaluation. The decision to use 4 less fresh assemblies is based on the early shutdown of Cycle 23. The reduced core burnup of Cycle 22 resulted in a lower fresh fuel requirement for Cycle 23. The reduced feed of fresh assemblies will result in improved fuel utilization and better economics.

Cycle 23 will also replace the existing control rods with new Enhanced Performance Rod Cluster Control Assemblies (EP-RCCAs). The replacement EP design is similar to the existing rods with the exception of a slight change in the length of the chrome plating on the outside of the absorber rods.

Summary of 50.59 Evaluation

The USAR Chapter 14 evaluations performed by NMC NAD and Westinghouse demonstrate that the Prairie Island Unit 2 Cycle 23 reload design and associated COLR do not result in the accepted safety limits for any accident being exceeded. The Cycle 23 design is consistent with the description of the core in the USAR. The core contains 121 fuel assemblies using a 14 x 14 fuel rod array, with 29 control rods in the same locations as described in the USAR. The only change from Cycle 22 is the distribution of new and used assemblies. This results in a redistribution of the isotopic distribution of the core that changes the core physics parameters of the reactor. The effect of these changes in the cycle physics parameters on cycle operation and accident analyses have been evaluated using NRC approved methods.

The accident analyses show that no design limits are exceeded during any analyzed transient for the cycle as designed. The cycle does not exceed any fuel burnup limits. Therefore the reload modification for Unit 2 Cycle 23 is safe and consistent with Prairie Island's current Licensing Basis.

50.59 Evaluation No. 1047 – Changes to Primary Chemistry Program Lithium and Hydrogen Limits

Description of Change

The proposed activity will revise chemistry procedures and the USAR such that the primary hydrogen limit is increased to 50 cubic centimeters per kilogram (cc/kg) of water and lithium concentrations of up to 5.0 parts per million (ppm) for the first 150 mega-watt day per metric ton of uranium (MWD/MTU) are acceptable during plant start-up following refueling. As the change will affect primary water chemistry, the change will affect the components in the Reactor Coolant System (RCS) including RCS piping and fuel cladding.

Currently, the USAR and chemistry procedures allow a maximum hydrogen concentration of 35 cc/kg of water (at standard temperature and pressure) and lithium concentration of 3.5 ppm. Further research in this area has determined that hydrogen concentrations up to 50 cc/kg of water have little effect to corrosion rate in primary systems. In addition, longer fuel cycles have resulted in increased beginning of cycle (BOC) boron concentrations such that in order to achieve the minimum pH of 6.9, lithium concentration would have to exceed the current 3.5 ppm limit. The procedure/document changes will implement revised limits for hydrogen and primary lithium control for beginning of fuel cycle operation, as described above and in the attached Westinghouse letter and in accordance with Electric Power Research Institute (EPRI) recommendations.

Summary of 50.59 Evaluation

The areas evaluated for the proposed change include:

- Effect on Operation of Safety Related Equipment The evaluation concluded that effects of elevated hydrogen and lithium would affect safety related equipment in the form of higher corrosion and initiation of Primary Water Stress Corrosion Cracking (PWSCC). Safety related SSC's in the emergency core cooling system (ECCS) would perform as designed. Corrosion and PWSCC are discussed in other sections of this evaluation.
- Crud Deposition within the Reactor Coolant System The evaluation concluded that the change will enable primary chemistry to maintain primary water at a minimum pH of 6.9. Maintaining pH at this level will minimize crud deposition within the RCS. Therefore there will be no adverse affect to crud deposition because of the change.
- Zirlo Corrosion The evaluation concluded that the increased hydrogen concentration will increase oxygen scavenging ability and not effect corrosion within the RCS. Changing the lithium concentration from 3.5 ppm to 5.0 ppm results in a negligible change in corrosion rate because the higher concentrations

are at the beginning of the cycle when boric acid offsets the corrosive effects of lithium, the change results in a minor increase in overall lithium exposure, and the use of Zirlo clad fuel is less susceptible to corrosion than previously used Zircaloy-4. Coupon test described in the EPRI guidelines have shown water <10 ppm lithium and >100 ppm boric acid has had negligible effects on Zircaloy corrosion.

- Primary Coolant Stress Corrosion Cracking (PWSCC) The evaluation concluded that the increase in primary water hydrogen and lithium concentration limits will minimally increase initiation and growth of PWSCC on RCS components. In comparison to temperature and stress, water chemistry is less significant to affecting PWSCC. In addition, studies referenced in the EPRI guidelines indicate that lithium concentrations of this magnitude (3-6 ppm) do not cause a detrimental effect. Routine inspections would find indications of PWSCC in susceptible materials prior to gross failure as indication and growth rate is slow, especially in low-duty reactors. In addition, new steam generators for Unit 1 and planned reactor vessel head replacement will replace old components with those less susceptible to PWSCC.
- Tritium Production The evaluation concluded that tritium production would be slightly higher as a result of the change. However the increase to offsite dose is negligible. Therefore the change is acceptable.
- USAR Chapter 14 Events The evaluation concluded that the change will not result in any new or increase of already analyzed accidents as described in the USAR. The additional tritium would not increase the consequences of a LOCA accident because tritium is not included as a source term in offsite or control room dose in accident scenarios. Therefore the change will not affect the accident analysis.

In summary, the evaluation concluded that the change will not adversely affect SSC's credited in the safety analysis and operation will remain within the licensing basis.

Note: The following summaries are for evaluations that were initiated prior to last major revision to 10 CFR 50.59, which was completed prior to the last update. However, these Evaluations were not previously reported because they were not closed until after the last update. Thus, instead of being called 50.59 Evaluations, they were called Safety Evaluations (and are referred to as such below.)

Safety Evaluation No. 336 – Erosion of Steam Generator Feedwater Thermal Sleeves

Description of Change
The purpose of this safety evaluation is to address the continued structural integrity of the steam generator feedwater nozzles with respect to the potential for and consequences of continued thermal sleeve erosion. Specifically, the potential for cracking in the feedwater nozzle bore, knuckle, and face regions is assessed. This safety evaluation concludes that the thermal sleeve erosion and resulting bypass flow will have no effect on the performance of the steam generator, and that feedwater nozzle integrity will be maintained through a future service period of 100 days of mode

2/mode 3 operation (i.e., when auxiliary feedwater is added to the steam generators during conditions such as hot standby or low power operation). Over the past four fuel cycles, Unit 1 has averaged about five days of auxiliary feedwater addition per cycle, and Unit 2 has averaged four days per cycle. If these trends continue, feedwater nozzle integrity should be maintained through at least the next three fuel cycles.

Summary of Evaluation
The structural integrity of the steam generator feedwater nozzles and thermal sleeves with respect to continued thermal sleeve erosion and potential cracking of the nozzle knuckle and bore regions, has been evaluated. Although cracking is not predicted, if it were to initiate, feedwater nozzle integrity will be maintained through a future service period of 100 days of mode 2/mode 3 operation (i.e., when auxiliary feedwater is added to the steam generators during conditions such as hot standby or low power operation). Based on the average period of auxiliary feedwater addition per cycle, over the last four years at either Prairie Island Unit 1 or Unit 2, feedwater nozzle integrity should be maintained through at least the next three fuel cycles. This safety evaluation concludes that the performance of the steam generator will be unaffected by the condition of the eroded thermal sleeves.

Based on this evaluation, it is concluded that the feedwater nozzle thermal sleeve erosion does not represent an unreviewed safety question, as defined in 10CFR50.59, and will not involve a change to plant technical specifications.

Safety Evaluation No. 345 – ASME Section XI Testing of Diesel Engine Generator and Auxiliary Support Systems with Diesel Fuel

Description of Change
Diesel support systems should be classified as equivalent to Quality Group C, and not Code Class 3. The diesel support systems are presently classified Code Class 3. Based upon RG 1.26, USAR Section 14, and GL 89-04, the diesel support systems will remain classified safety related, however they will not be code classified.

Summary of Evaluation
The change in code designation allows increase or decrease in the frequency and extent of testing of diesel support systems based on technical merit and equipment reliability, but will not degrade or prevent actions described or assumed in the USAR. Existing Inservice Testing (IST) requirements will not be discontinued or changed without evaluation of the overall effect of these changes on system performance such that accidents are less likely to occur as a result of flexibility in testing requirements. Systems and components reliability will be evaluated before any or increase or decrease in testing occurs, therefore no increase in probability of occurrence of a new malfunction will occur. Malfunctions and failure history are inputs to the Reliability Centered Maintenance process. Inspections and tests are devised to detect malfunctions and change in system, structure, and component (SSC) performance; therefore the change will not increase the consequence of a malfunction unless the tests and inspections are destructive in nature. TS 4.2 requires inservice testing of code class 1, 2, and 3 SSCs. Removing a SSC from the IST Program and performing tests to demonstrate the SSC will perform satisfactorily in service will not reduce the margin of safety as defined in the basis for TS 4.2.

Safety Evaluation No. 373, Addendum 1, Revision 1 – Removal of Containment Sump B to RHR Valve Pressure Locking/Thermal Binding Cycling Requirements

Description of Change
During the middle of March 1995 additional information was made available to the industry from utilities and also the NRC. A utility declared their recirculation sump valves inoperable based on new calculations which they had performed which evaluated their recirculation sump valve operability taking into effect the valve bonnet heatup/pressurization due to the accident environment (Sump accident fluid temps and ambient room heatup). Based on this information and conversations which had taken place between the NRC and Prairie Island, a further review of Prairie Island's SE 373 was done. Upon this review, an error was discovered in the calculation that is the basis for the conclusion of SE 373 addendum 1. The error found did not alter the conclusion of the safety evaluation, but did require correcting. It was decided that a revision to the addendum should be written to correct the error and also include the additional guidance given to us in a NRC Temporary Instruction (TI 2515/129).

Summary of Evaluation
The Sump B to RHR Pump suction valves are not susceptible to thermal binding and will perform their design function under pressure locking scenarios. The conclusions are valid for an interim period based on the fact that the NRC has accepted the calculation methodology used only in the interim. Prairie Island during the Unit 2 May 1995 outage will perform modifications to the pump-side Sump B valves in order to justify long term operability of the valves.

In addition, procedure changes have been implemented to verify the assumptions used in this addendum are justifiable. A temporary memo has been written to cycle the sump-side Sump B valves prior to leaving cold shutdown. This change has been performed as a verification to any maintenance, or surveillance operation which may have introduced water into the line between the pump-side and sump-side Sump B valves. Operability of the sump-side valve is based on air in the bonnet.

Temporary memos have been written for the EOPs, which require venting of the bonnets prior to going onto recirculation to relieve any pressure which may have built up in the valve bonnets. The calculation demonstrates that a slight increase in bonnet fluid temperature develops a large pressure in the bonnets due to the incompressibility of water in a constant volume. The calculations also show that the capability of the current motor operators is sufficient to overcome this force, for the interim period. The NRC has not accepted this for long term operation, such that bonnet vents will be installed and the bonnets must be vented prior to going on recirculation.

Design Change 87Y775 - Main Generator RTD and Fiber Optic Vibration Instrument (Unit 1)

Summary of Change
Design Change 87Y775 installed additional generator resistance temperature detectors (RTD's) in Unit 1 generator gas passages and added fiber optic vibration sensors to the

generator field windings end turns to detect loose windings and prevent generator failures. Half of the installed RTD's were terminated at the existing generator terminal board and 6 existing imbedded winding RTD's were disconnected. New fiber optic vibration electronic equipment was installed and connected to the new sensors.

Summary of Basis of Determination
This modification does not create the possibility of an accident or malfunction of equipment important to safety of a different type than evaluated previously in the USAR nor increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the USAR since the additional sensors are similar to the original design sensors and are mounted in a similar manner as the original sensors. Both the sensor design and mounting are performed as designed by the original equipment manufacturer.

This modification does not increase the consequences of any accident or malfunction of equipment important to safety previously analyzed in the USAR since the equipment affected by this change is not credited in any accident analysis and this equipment is not important to safety.

This modification does not reduce the margin of safety as defined in the bases for any Technical Specification since the generator is not addressed in any Technical Specifications of bases.

Design Change 91L268 - Fire Zone 67 and 73 Wiring Change

Summary of Change
Design Change 91L268 modified several fire detection circuits due to cable failures. The fire detectors in fire detection zone #73, which covers the two Deepwell Pumphouses, were abandoned in place due to failures of the cables between the detectors and Fire Detection Panel 126. This change is acceptable because the equipment located in Deepwell Pumphouse 1 and 2 are not Tech Spec-related nor required for safe shutdown of the plant.

The fire detectors in fire detection zone #67, which covers the Cooling Tower Pumphouse, were reconnected to fire detection zone #71, which covers the Cooling Tower Equipment House, due to a failed cable between the Cooling Tower Pumphouse and Fire Detection Panel 126. Fire detection zone #67 was labeled spare and fire detection zone #71 was relabeled to include the Cooling Tower Pumphouse. This change is acceptable because fire detection for zone 67 was restored to operable by expanding the scope of fire detection zone #71. This change expanded the scope of fire detection zone #71 but did not change its function.

Summary of Basis of Determination
This modification will not increase the consequences or probability of occurrence of an accident previously identified in the SAR or of a malfunction of equipment important to safety previously evaluated in the SAR since the equipment in the locations affected by these fire detection changes are neither credited in any accident analysis nor important to safety. These changes also will not increase the probability of failure of the fire detection system since this system will continue to function within its design. The

function of the fire detection system is not changed by this modification. This modification will not create the possibility of an accident of a different type than any previously evaluated in the SAR nor create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the SAR since the equipment in the locations affected by these fire detection changes are neither credited in any accident evaluation nor important to safety.

Design Change 96CL02 - Emergency Intake CL Line Chemical Addition Line

Description of Change

The purpose of this modification is to install a permanent pipe line that can be used to chemically treat the Cooling Water System Emergency Intake Line. The line is subject to potential accumulation of zebra mussels. The ability to chemically treat the line will minimize this potential vulnerability.

Summary of 50.59 Evaluation

The emergency intake line and crib are not accident initiators. Thus, there is no increase in frequency of any accident or creation of an accident of a different type. The chemical addition line has no affect on the integrity of the emergency intake line and crib. The attachment of the chemical addition line has no affect on the performance of the emergency intake line or crib during or after a design bases earthquake. Thus, there is no change to the consequences of any accident analyses or due to any equipment malfunctions. In addition, these changes do not increase the likelihood for equipment malfunction nor create an equipment malfunction with a different result.

ATTACHMENT 2

CHANGES TO REGULATORY COMMITMENTS

Regulatory Commitment Change 03-04 - Emergency Intake Crib Inspection Frequency

In response to Generic Letter 89-13, NMC is committed to inspecting the Emergency Intake crib every five years. The change allows the crib to be inspected at the same five-year interval, but +/- 25%. Previous inspections indicate that an additional flexibility in the frequency is warranted.

Regulatory Commitment Change 04-04 - Revise Commitment for Single Inverter OOS

Change the commitment to note that, in lieu of any specific time limit, when a single inverter is out of service, the Maintenance Rule (a)(4) program is used to assess and manage the risk associated with this plant configuration.

Regulatory Commitment Change 05-01 - Undervessel Inspection Frequency

Added to the commitment (to perform an undervessel inspection during each refueling outage) is an allowance to perform the undervessel exam within one month of the actual refueling outage (should the opportunity arise).

Regulatory Commitment Change 05-02 - Add Flexibility to CFCU Retest Frequency

Heat Exchanger Retests - In response to Generic Letter 89-13, implement a periodic program to retest for degraded performance of the large Cooling Water cooled safety - related heat exchangers. The minimum retest frequency as listed in Generic Letter 89-13 is 5 years. Change would allow CFCU retests on 5 year +/- 20% frequency.