WCAP-11067

PRAIRIE ISLAND UNIT 1

REACTOR VESSEL UPPER INTERNALS REPLACEMENT

SAFETY EVALUATION

K. J. Voytell, Jr.

APPROVED BY: C W Hirst

C. W. Hirst, Manager
Reactor Coolant Systems Components Licensing

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EXECUTIVE SUMMARY

Westinghouse has been contracted by Northern States Power Company to supply a newly designed upper internals assembly package for their Prairie Island Unit 1 Nuclear Station. The proposed package will provide many state-of-the-art design improvements over the current design. These include an improved Rod Control Cluster (RCC) guide tube design featuring an advanced support pin design, a simplified fully-machined upper support assembly with fewer parts and weld joints, a refined upper support column design, elimination of flow mixers, and a thicker upper core plate. An overall significant reduction in the quantity of parts and components should provide improved reliability. Additionally, replacement of the existing upper internals in lieu of only support pin or guide tube replacement will result in a cost savings while providing for a reduction in outage time required to make the modifications.

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1.0 BACKGROUND AND INTRODUCTION

Several Westinghouse operating plants have experienced failures due to stress corrosion cracking of the guide tube support pins manufactured from Inconel X750 material. In reaction to these failures, Westinghouse undertook a comprehensive investigation program. This program confirmed that these support pin failures were occurring in Inconel X750 material that had a solution heat treatment of less than [ ]^{a,c,e} Subsequent corrective action to date has included replacement of support pins in 5 operating plants and 50 non-operating plants. The replacement pins of improved X750 material incorporated various new design features, including a higher temperature solution heat treatment of the material. Test results showed that this enhanced design will reduce the susceptibility to stress corrosion cracking. The geometric design changes were minimized so that the replacement pin was interchangeable, and could be incorporated with an acceptable level of difficulty into existing guide tubes (some of which were in operating plants).

Additional design margin can be achieved by replacement of the upper internals assembly complete with a modified guide tube design. This will allow additional design enhancements to the support pin that were not practical to incorporate into existing guide tubes.

Finally, replacement of the existing upper internals in lieu of only support pin or guide tube replacement will result in a cost savings while providing for a reduction in outage time required to make the modifications.

The evaluation to follow will provide adequate assurance that the various design changes and modifications which compose the upper internals assembly package will not affect the safety margin of the Prairie Island Unit 1 Nuclear Station.

2.0 DESCRIPTION OF COMPONENTS AFFECTED BY UPPER INTERNALS REPLACEMENT

The following provides brief descriptions of the components affected by the upper internals replacement and their functions.

2.1 UPPER CORE SUPPORT ASSEMBLY

The upper core support assembly provides the vertical and lateral restraint and lateral alignment to the top of the core through its primary components (the upper support subassembly, support columns, and the upper core plate) and its interface with the reactor vessel. The assembly also provides the support for the internals structures, such as the instrumentation conduit and supports, and reactor control rod guide tubes.

The upper support subassembly, which is supported on the outer edges, transfers the loading of the upper core support assembly to the reactor vessel. Keyways with customized inserts to maintain required gaps are located in the outer edges of the subassembly to provide the upper core support assembly to reactor vessel to lower core support assembly alignment and to limit any transverse or rotational movement of the subassembly. There are penetrations through the subassembly for spray nozzles which allow limited flow into the reactor vessel head area. The support columns transfer vertical and lateral loads to the upper support subassembly and support the upper core plate vertically.

The upper core plate, which is attached to the bottom of the upper support columns, forms the upper periphery of the core. The upper core plate transfers core loading to the support columns, and when in place within the reactor vessel, supplies a preload to the fuel matrix. The plate is perforated to allow coolant flow while maintaining a uniform velocity profile. The underside of the plate contains the upper fuel pins which engage the top of the fuel assemblies. [

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2.2 UPPER INSTRUMENTATION CONDUIT AND SUPPORTS

The conduit and supports provided in the upper core support assembly function to provide a passageway, cross-flow support, and end stops for thermocouples which are inserted into the top of the core support structure. These internal structures are attached to the upper support subassembly and are inside of the support columns with the thermocouple end stops protruding into the measurement areas.

2.3 REACTOR CONTROL ROD GUIDE TUBES

The reactor control rod guide tubes perform the following functions:

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3.0 REACTOR VESSEL REPLACEMENT UPPER INTERNALS DESCRIPTION

3.1 UPPER INTERNALS ASSEMBLY DESIGN

The replacement upper internals assembly will incorporate state-of-the-art technological design enhancements. The assembly, shown in Figure 3-1, will feature a [

.]a,c,e Of prime importance is the incorporation of an advanced non-welded type 316 stainless steel guide tube support pin design. The following discussions describe the replacement upper internals assembly design and highlight component changes and modifications.

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REPLACEMENT UPPER INTERNALS ASSEMBLY

FIGURE 3-1

3.1.1 Upper Support Assembly

The replacement upper support design is shown in Figure 3-1. The design is commonly referred to as an "inverted top hat" due to its characteristic shape. This type of design is featured in all of the latest Westinghouse plants, including two-, three-, and four-loop reactor internals. The design features a support flange, a full-circumferential skirt, and a thick support plate. This all-machined design is much simpler than the existing fabricated deep beam upper support pictured in Figure 3-2, with its many parts and weld joints. The new design allows a reduction in the overall length of the lower sections of the RCC guide tubes.

3.1.2 Upper Support Columns

The replacement upper support column is pictured in Figure 3-1. The design includes a solid, non-slotted body, and a cruciform style base which attaches to the upper core plate with [ ]^{a,c,e} diameter bolts. The top end of the column is flanged and is attached to the upper support plate with [ ]^{a,c,e} bolts. Each support column houses a single thermocouple which protrudes from its base directly above the core. Thirty-six identical support columns will be included in the replacement upper internals design. Thirty-nine thermocouples will be maintained in the replacement upper internals design. A typical existing support column is shown in Figure 3-1. The simplification of the design is clear, with a significant reduction in the number of parts.

3.1.3 Elimination of Flow Mixers

Westinghouse experience with operating plants has demonstrated that the use of flow mixers above the core to mix flows of varying temperatures is not essential to accurate measurement of core performance. All of the latest Westinghouse reactor internals designs exclude this extraneous feature. The proposed upper internals design does not require flow mixers. As mentioned above, the thermocouples will protrude from the lower end upper support columns directly above the core.

3.1.4 14 x 14 RCC Guide Tubes

The current RCC guides at Prairie Island Unit 1 feature three sections: an upper, intermediate, and lower or "continuous" section. The sections bolt together as shown in Figure 3-3. [

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a,c,e

EXISTING UPPER INTERNALS ASSEMBLY

FIGURE 3-2

[ ] a,c,e

GUIDE TUBE ASSEMBLY

14 x 14

FIGURE 3-3

[ ] a,c,e

GUIDE TUBE ASSEMBLY

14 x 14 MODIFIED WITH 16 x 16 FEATURES

FIGURE 3-4

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]^{a,c,e}

3.1.5 Thermocouple Columns and Upper Instrumentation

The replacement upper internals design will maintain three thermocouple columns. The locations of the thermocouples will, to the extent practical, duplicate the existing design. The elimination of the flow mixers simplifies the upper instrumentation conduit layout. The conduits will run from each support column location to the thermocouple columns with intermittent support from brackets mounted to the upper support plate.

3.1.6 Upper Core Plate

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]^{a,c,e} The replacement core plate hole pattern is compatible with the current design.

3.2 UPPER INTERNALS ASSEMBLY CUSTOMIZATION

Due to the critical fit-up requirement between the upper and lower internals, customization of the upper internals was required. Westinghouse has utilized all available original manufacturing information in combination with newly acquired inspection data in determining the customization requirements for the upper internals. This has required the design of special gauging equipment to measure the orientations and sizes of the upper core plate alignment pins. The replacement internals have been appropriately customized to provide proper fit and clearances with the existing lower internals.

4.0 REACTOR PRESSURE VESSEL SYSTEMS ANALYSIS OBJECTIVES

4.1 HYDRAULIC ANALYSES

This section documents the hydraulic analyses which were performed to support the Prairie Island Unit 1 upper internals replacement. In the first subsection the guide tube loss coefficient is evaluated based on 16x16 data (Ref. 4.1.1), and the modified 14x14 guide tube characteristics. Also determined in this subsection is the core plate hole diameter needed to ensure that the flows through core plate holes without guide tubes are acceptably close to those through guide tubes. The second subsection deals with the reactor pressure vessel system hydraulic characteristics (pressure drops, etc.) which will be affected by the replacement upper internals. In the third subsection the normal operation hydraulic crossflow loads on the replacement guide tubes and support columns are calculated.

4.1.1 Guide Tube Hydraulic Loss/Core Plate Hole Sizing

4.1.1.1 Analysis Objectives

The purpose of this analysis is to determine: a) the hydraulic loss coefficient presented by the replacement guide tubes to the flow between the core exit and the outlet plenum, and b) the non-guide tube core plate hole diameter needed to ensure that the flow rates in guide tube and non-guide tube holes are sufficiently close to each other.

4.1.2 System Hydraulic Analysis

4.1.2.1 Analysis Objectives

The replacement upper internals are affected by and potentially affect the reactor pressure vessel system hydraulic characteristics. These characteristics are important in determining the fluid forces, pressure drops acting on the reactor internals, and the pressure drops and fluid velocities at various points within the reactor vessel. The purpose of this analysis is to present the results of the reactor pressure vessel system hydraulic analysis with the replacement upper internals installed.

4.1.3 Upper Internals Hydraulic Loads

4.1.3.1 Analysis Objectives

The purpose of this analysis is to present the results of calculations performed for the steady-state crossflow hydraulic loads on the most highly-loaded guide tubes and support columns. These guide tubes and support pins will be the ones nearest the outlet nozzles, where fluid velocities are highest. This is done for mechanical design, hot pump overspeed and cold full flow conditions, and for the thimble-plugs-absent and thimble-plugs-present cases.

4.2 FLOW-INDUCED VIBRATION ANALYSIS

4.2.1 Analysis Objectives

Flow-induced vibrations of pressurized water reactor internals have been studied at Westinghouse over a number of years. The objective of these studies is to assure structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale-model tests, bench tests of components, and various analytical investigations. The results of scale-model and in-plant tests indicate that the vibrational behavior of two-, three-, and four-loop plants is essentially similar; the results obtained from each of the tests complement one another and make possible a better understanding of the flow-induced vibration phenomena.

One issue that was addressed in the replacement of the Prairie Island upper internals is whether or not the vibration characteristics of the system are changed significantly, how they are changed, and whether these changes are acceptable. A vibration assessment to address these questions has been performed and is detailed in (Ref. 4.2.1). The purpose of this assessment was to demonstrate that the proposed replacement upper internals assembly is similar in performance to that of existing upper internals in service.

There were basically two aspects of this assessment that were considered: a) the effect of the replacement of upper internals on the vibration characteristics of the lower internals, and b) the vibration characteristics of the replacement of upper internals.

4.3 ROD CONTROL CLUSTER ASSEMBLY (RCCA) WEAR ANALYSIS

4.3.1 Analysis Objectives

Fretting wear in RCCAs is caused by the frictional forces resulting from the relative vibrational motion between the rods and the guide cards. Obtaining an accurate estimate of the wear rate is difficult because of the large number of parameters involved. Many parameters, such as the initial profile of the given rod relative to the cards that guide it, are unknown and will vary from one rod to the next. For this reason, an absolute estimate of fretting wear was not attempted. Instead, a comparative analysis of the RCCA wear in the replacement guide tube vs. the existing guide tube was performed. [

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The RCCA wear due to stepping was also considered in this evaluation. This type of wear can be assessed on a relative basis by comparing the hydrodynamic loads which hold the rodlets against the sheaths and split tubes. [

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4.4 CONTROL ROD DROP TIME ANALYSIS

4.4.1 Analysis Objectives

The purpose of control rod drop time analysis is to provide assurance that the Control Rod Drive Mechanism (CRDM) scram times for Prairie Island Unit 1, with replacement internals and with a mix of Exxon and Westinghouse fuel, will be within acceptable limits. The final verification will be the start-up rod drop time tests.

4.5 QUALITATIVE COMPARISON TO EXISTING INTERNALS ANALYSIS

4.5.1 Analysis Objectives

A qualitative analysis was conducted to better define the structural differences between the proposed replacement upper internals and those of the corresponding original upper internals. The objective of this study is to demonstrate that the proposed upper internals are either equivalent or superior structurally and functionally to the existing internals.

5.0 ACCIDENT ANALYSIS, LOSS OF COOLANT ACCIDENT (LOCA)

5.1 LOCA HYDRAULIC FORCES ANALYSIS

5.1.1 Analysis Objectives

The purpose of this analysis is to determine the dynamic response of the Prairie Island Unit 1 Reactor Pressure Vessel (RPV) with replaced upper internals to four postulated pipe ruptures. The ruptures considered include: a) RPV inlet nozzle break b) RPV outlet nozzle break c) steam generator inlet break, and d) pump outlet break. Analysis objectives include determination of the displacements of the RPV, impact loads, as well as internals components loadings.

5.2 SMALL BREAK EMERGENCY CORE COOLING SYSTEM (ECCS) LOCA ANALYSIS (CYCLE 11)

5.2.1 Analysis Objectives

The NOTRUMP (Ref. 5.2.1), and LOCTA-IV (Ref. 5.2.2) computer codes are utilized for a spectrum of small break sizes. These codes are incorporated in the approved Westinghouse ECCS Small Break Evaluation Model (Ref. 5.2.3) developed to determine the Reactor Coolant System (RCS) response to design basis small break LOCAs and to address the Nuclear Regulatory Commission (NRC) concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break LOCA in Westinghouse-Designed Operating Plants." Considered in the analysis are the proposed hardware enhancements, i.e., new upper internals package and thimble plug removal, as well as increased levels of F_Q and steam generator tube plugging. The objective of the small break ECCS LOCA analysis is to demonstrate that Prairie Island Unit 1 conforms to the requirements of Appendix K and 10CFR50.46 for small break LOCA.

5.3 LARGE BREAK ECCS LOCA ANALYSIS (CYCLE 11)

5.3.1 Analysis Objectives

The SATAN-VI (Ref. 5.3.1), WREFLOOD (Ref. 5.3.2), COCO (Ref. 5.3.3), and LOCTA-IV (Ref. 5.2.2) computer codes are utilized for a spectrum of large break sizes. These codes are incorporated in the approved Westinghouse ECCS Large Break Evaluation Model (Ref. 5.3.4) developed to determine the RCS response to design basis large break LOCAs. Considered in the analysis are the proposed hardware enhancements, i.e., new upper internals package and thimble plug removal. The objective of the large break ECCS LOCA analysis is to demonstrate that Prairie Island Unit 1 conforms to the requirements of Appendix K and 10CFR50.46 for large break LOCA.

5.4 CRDM LOCA AND SEISMIC ANALYSIS

5.4.1 Analysis Objectives

Several dynamic analyses of the control rod drive mechanisms (CRDMs) were undertaken for Prairie Island Units 1 and 2. The analyses performed considered the effects of Upset (DBE + D.W.) and Faulted loading ($(SSE^2 + LOCA^2)^{1/2}$ + DW) conditions on the structural integrity. Margins for the loads on CRDM pressure boundary components were calculated. The purpose of these analyses is to demonstrate that the structural integrity of the CRDMS is preserved through various postulated accidents.

6.0 ACCIDENT ANALYSIS, NON-LOCA

6.1 NON-LOCA ANALYSIS

Under the terms of the fuel contract Westinghouse will perform a review of the Northern States Power Company generated document:

PRAIRIE ISLAND UNIT I, CYCLE 11
FINAL RELOAD DESIGN REPORT
(RELOAD SAFETY EVALUATION)
NSPNAD-8510P

Westinghouse comments are to appear under separate cover.

6.2 SEISMIC ANALYSIS

6.2.1 Analysis Objectives

The primary objective of the analysis is to determine a realistic seismic response of of the RPV and internals system in order to demonstrate that the structural integrity is maintained for all reactor internals components.

Additionally, the analysis is to obtain absolute displacements of the upper and lower core plate and the barrel.

7.0 SAFETY EVALUATION

7.1 HYDRAULIC EVALUATION

7.1.1 Guide Tube Hydraulic Loss/Core Plate Hole Sizing Evaluation

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7.1.2 System Hydraulic Evaluation

Simplified analytical techniques were used to calculate velocities, pressure drops, and forces in the reactor vessel using one-dimensional equations for continuity and mechanical energy.

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Reactor internals pressure drops and lift forces were calculated for the four flow conditions described below:

a) Thermal Design Flow: A conservatively low flow used to calculate steady-state and transient thermal performance of reactor internals.

b) Mechanical Design Flow: A conservatively high flow used to calculate steady-state and vibratory hydraulic loads on reactor internals.

c) Hot Pump Overspeed: A condition in which the pump rotational speed increases, thereby producing higher plant flow rates tha normal.

d) Cold Pump Overspeed: The plant flow rate at cold, zero power conditions.

At thermal design, mechanical design, and hot pump overspeed flow rates, the reactor plant is at full power with T_{in} = 535.5°F. At cold full flow, reactor power is zero and T_{in} = 70°F. Pressure drop and lift force distributions were calculated for two cases:

a) Thimble plugs present.

b) Thimble plugs removed.

In the latter case the bypass flow was increased from [ ]^{a,c,e} to [ ]^{a,c,e} and the cold full flow and hot pump overspeed flow rates were increased by [ ].^{a,c,e}

a,c,e

HYDRAULIC FLOW PATHS

FIGURE 7-1

No lift forces were calculated for the thermal design case, because the lift forces at thermal design flow will always be less than those at mechanical design flow and will therefore be non-limiting.

7.1.3 Upper Internals Hydraulic Loads Evaluation

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7.2 FLOW-INDUCED VIBRATION EVALUATION

7.2.1 Lower Internals

It has been demonstrated through both experimental and analytical results that the various mode frequencies and displacement amplitudes of the core barrel are very much the same (Ref. 7.2.1, Section 4.5.1), independent of whether or not the upper internals are present. As a consequence it can be reasonably concluded that the vibration characteristics of the core barrel are relatively unaffected by the upper internals, even in the extreme case where they are absent. Because the Prairie Island modification involves replacement, not removal, of the upper internals, its effect on core barrel vibration is expected to be negligible.

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FIGURE 7-2

a,c,e

FIGURE 7-3

a,c,e

FIGURE 7-4

a,c,e

FIGURE 7-5

7.2.2 Upper Internals

7.2.2.1 Guide Tubes

The excitation forces acting on the guide tubes fall into two categories:

a) Crossflow-induced.

b) Pump-induced.

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7.2.2.2 Support Columns

The Prairie Island Unit 1 replacement support columns are the same design used in the newer Westinghouse two loop plants. Therefore, it is sufficient to demonstrate that the vibrational loads acting on the Prairie Island Unit 1 replacement support columns are less than or equal to those to which the standard two-loop support columns were designed. This comparison is favorable, because the standard two-loop support columns were designed to pump-induced loads which are higher than those which occur in a two-loop plant.

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The Prairie Island Unit 1 total vibrational support column loads are well below the original design loads. It can, therefore, be concluded that the vibration amplitudes of the support columns in Prairie Island Unit 1 will be well below acceptable levels.

7.2.2.3 Upper Support Plate

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7.3 RCCA WEAR EVALUATION

The RCCA wear analysis has employed models to analyze the effects of what are considered to be the most probable causes of fretting wear. Figure 7-6 is an idealization of the possible fretting wear mechanisms. For illustrative purposes, only a rod-sheath combination is shown, but the concepts apply to split tubes as well.

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FRETTING WEAR DUE TO GUIDE TUBE VIBRATION PARALLEL AND PERPENDICULAR TO RODLET PRESSURE LOADING

FIGURE 7-6

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7.4 CRDM DROP TIME EVALUATION

With the required parameters established for the fuel, internals, and the balance of the reactor, actual rod drop times can be estimated. To assure conservatism, the analyses for operating conditions assumed a Mechanical Design Flow (MDF) condition of 103,300 gal/min/loop, and conservatively high values of mechanical friction forces compared to those obtained from actual tests.

The results of the various analyses (Ref. 7.2.1, Section 6), indicate that the drop times fall well within the Technical Specification limit of 1.8 sec. To achieve a very high level of confidence that this requirement will be met during plant operation, parametric studies were performed. These studies varied individual parameters, holding others at nominal values. Hence, when in-plant verification tests are performed, there is adequate assurance that the 1.8 second limit will be met.

7.5 STRESS EVALUATION

The Prairie Island Unit 1 reactor internals components were designed prior to the 1974 ASME Code (with the addition of subsection NG), and consequently a design specification was not a requirement. The replacement upper internals and thermocouple column assembly are designed to the generic design specifications (Refs. 7.5.1, Section 6.0, 7.5.2, Section 6.0, respectively), and constructed in accordance with the site specific Prairie Island Unit 1 design specification.

7.5.1 Upper Internals

A structural and functional comparison study was conducted between the upper internals replacement components and the existing upper internals. From the results (Ref. 7.5.3), of this study it can be concluded that the replacement components are either equivalent to or an improvement over the existing upper internals. The core support components; upper support plate, flange, skirt, upper support column and fasteners, upper core plate, and core plate inserts meet the ASME Boiler and Pressure Vessel Code requirements for Section 3, Division 1, Sub-Section NG 1974 Edition, 1976 Summer Addenda. The replacement internals components are constructed in such a way as to not adversely effect the integrity of the core support structure. Documentation of compliance with the design specification functional requirements, namely the displacement allowables during faulted conditions and margins of safety for normal (level A) and normal plus upset conditions (level A+B) are included in the design report, (Ref. 7.5.1).

7.5.2 Thermocouple Column Assembly

The thermocouple column assembly satisfies the applicable portion of the ASME Boiler and Pressure Vessel Code, Section 3, Division 1, 1974 Edition, 1974 Summer Addenda. Documentation of compliance with the design specification functional requirements and margins of safety are included in the design report, (Ref. 7.5.2).

7.6 LOCA HYDRAULIC FORCES EVALUATION

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7.6.1 Reactor Internals Impact Loads

The dynamic impact loads for the reactor internals are the primary interfacing loads between the reactor internals components and the reactor pressure vessel. [

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The reactor core is composed of 121 fuel assemblies. Each assembly is horizontally positioned by two fuel pins on the lower core plate, and two on the upper core plate. During the closure of the reactor head, the holddown spring for each fuel assembly is compressed, and hence, the entire core is restrained vertically.

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TABLE 7-7

SUMMARY OF RP/RI INTERFACE LOADS

RP Vessel/RI Interface

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a,c,e

FIGURE 7-8

7.7 SMALL BREAK ECCS LOCA EVALUATION (CYCLE 11)

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]a,c,e During the second period of uncovery a Peak Clad Temperature (PCT) of 1000°F occurs. This value is well below all Acceptance Criteria limits of 10CFR50.46 and is non-limiting in comparison to large break analysis results.

7.8 LARGE BREAK ECCS LOCA EVALUATION (CYCLE 11)

Of the three break size coefficients evaluated: C_D=0.4, C_D=0.6, and C_D=0.8, the C_D=0.4 break proved to be the limiting (highest PCT) case, with a PCT of 2098°F. The C_D=0.6 and C_D=0.8 yielded PCTs of 2000°F and 1999°F respectively. Current NRC restrictions require that a penalty be assessed and imposed for insufficient modeling of the Upper Plenum Injection (UPI). This penalty was assessed to be 78°F for the limiting case. [

]a,c,e Imposition of these penalties results in a final clad temperature of 2186°F which is below the 2200°F Acceptance Criteria limit established by Appendix K of 10CFR50.46. The Revised PAD Thermal Safety Model (Ref. 7.8.1), was used to generate fuel input parameters for the analysis. [

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7.9 CRDM LOCA AND SEISMIC EVALUATION

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TABLE 7-9

DESIGN EARTHQUAKE (OBE) RESPONSE SPECTRA FOR
CRDM ANALYSIS, 5% EQUIPMENT DAMPING

HORIZONTAL

Frequency (hz)	Acceleration (G)	Frequency (hz)	Acceleration (G)
1.0	0.083	5.6	0.30
1.5	0.173	5.8	0.29
1.7	0.235	6.0	0.27
1.9	0.351	6.2	0.25
2.0	0.395	6.8	0.23
2.2	0.610	7.0	0.22
2.4	1.020	8.0	0.22
2.6	1.02	9.0	0.23
2.8	1.09	9.5	0.24
3.0	1.09	10.0	0.30
3.4	1.05	11.0	0.34
3.6	0.91	12.0	0.34
3.8	0.675	13.0	0.38
4.0	0.63	15.0	0.38
4.2	0.55	16.0	0.37
4.4	0.50	17.0	0.34
4.6	0.42	22.0	0.29
4.8	0.36	36.0	0.16
5.0	0.30	80.0	0.14

TABLE 7-10

DESIGN EARTHQUAKE (OBE) RESPONSE SPECTRA FOR
CRDM ANALYSIS, 5% EQUIPMENT DAMPING

VERTICAL

Frequency (hz)	Acceleration (G)	Frequency (hz)	Acceleration (G)
1.0	0.04	3.08	0.25
1.33	0.04	3.33	0.265
1.38	0.041	3.64	0.275
1.43	0.045	4.00	0.27
1.48	0.048	4.20	0.26
1.54	0.053	4.44	0.22
1.60	0.060	5.00	0.16
1.67	0.069	5.71	0.13
1.74	0.079	6.67	0.10
1.82	0.087	8.00	0.085
1.91	0.100	10.0	0.075
2.0	0.115	13.33	0.068
2.1	0.13	20.0	0.062
2.22	0.14	40.0	0.061
2.35	0.16	80.0	0.60
2.50	0.18		
2.67	0.20		
2.86	0.23		

]a,c,e

Component loadings are compared to allowable loadings for the OBE and Faulted conditions. Note that the Faulted condition is defined as follows:

$$\text{Faulted} = (SSE^2 + LOCA^2)^{1/2} + DW$$

Table 7-11 shows the maximum bending moments along the latch housing (LH) and rod travel housing (RTH) of the CRDM, for the OBE, SSE and LOCA events and allowables. Allowables were determined by detailed pressure boundary analyses (Ref. 7.9.4) on the CRDM components, in accordance with the ASME code, Section III. By the actual upset and faulted loadings being less than these allowables, the ASME code stress allowables are met.

All components of the pressure boundary of the control rod drive mechanisms demonstrate structural adequacy for the postulated seismic and LOCA conditions.

7.10 NON-LOCA EVALUATION

Please refer to section 6.1 of this evaluation.

7.11 SEISMIC EVALUATION

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TABLE 7-11

CRDM MAXIMUM BENDING MOMENT (in-lbs)

a,c,e

TABLE 7-12

RESPONSE ACCELERATION SPECTRA TABULATION
(HORIZONTAL ACCELERATION)
EARTHQUAKE IN N-S OR E-W DIRECTION
MASS POINT 17, 17A
DAMPING RATIO - 0.005, and 0.010

Period Sec.	Damping Ratio 0.005	0.010
0.000	0.150	0.120
0.025	0.180	0.150
0.038	0.225	0.180
0.050	0.450	0.300
0.063	0.630	0.540
0.075	0.640	0.550
0.088	0.630	0.540
0.100	0.450	0.400
0.113	0.270	0.220
0.125	0.250	0.210
0.150	0.270	0.250
0.175	0.400	0.350
0.200	0.600	0.450
0.225	0.850	0.650
0.250	1.150	0.900
0.275	1.600	1.280
0.300	2.050	1.700
0.325	2.600	2.220
0.338	2.775	2.250
0.350	2.790	2.265
0.363	2.790	2.255
0.375	2.770	2.240
0.400	2.300	1.850
0.425	1.750	1.400
0.450	1.290	0.900
0.475	0.850	0.670
0.500	0.650	0.550
0.525	0.510	0.450
0.550	0.420	0.380
0.575	0.380	0.320
0.600	0.340	0.280
0.625	0.300	0.255
0.650	0.270	0.230
0.675	0.230	0.210
0.700	0.210	0.195
0.725	0.190	0.180
0.750	0.175	0.165
0.775	0.165	0.150
0.800	0.150	0.145
0.825	0.147	0.147
0.850	0.143	0.143
0.875	0.132	0.132
0.900	0.127	0.127
0.925	0.125	0.125
0.950	0.118	0.107
0.975	0.113	0.113
1.000	0.110	0.110
1.250	0.100	0.100
1.500	0.080	0.080
1.750	0.060	0.060
2.000	0.040	0.040

]a,c,e

TABLE 7-13

RESPONSE ACCELERATION SPECTRA TABULATION
(VERTICAL ACCELERATION)
EARTHQUAKE IN N-S OR E-W DIRECTION
MASS POINTS 4, 4A, 6, 6A, 9, 9A, 10, 10A
14-20, 14A-20A, 21, 24, 25, 28-32, 33, 33A, 41, 42
DAMPING RATIO - 0.005, 0.010, 0.020, and 0.050

Period Sec.	Damping Ratio 0.005	0.010	0.020	0.050
0.000	0.060	0.060	0.060	0.060
0.025	0.062	0.061	0.061	0.061
0.050	0.072	0.065	0.063	0.062
0.075	0.085	0.080	0.072	0.068
0.100	0.100	0.095	0.085	0.075
0.125	0.120	0.110	0.100	0.085
0.150	0.145	0.138	0.120	0.100
0.175	0.180	0.165	0.150	0.130
0.200	0.230	0.210	0.190	0.160
0.225	0.760	0.460	0.270	0.220
0.238	0.785	0.635	0.430	0.260
0.250	0.788	0.635	0.435	0.270
0.275	0.780	0.625	0.440	0.275
0.300	0.760	0.605	0.435	0.265
0.325	0.730	0.580	0.420	0.250
0.350	0.660	0.520	0.380	0.230
0.375	0.560	0.420	0.300	0.200
0.400	0.420	0.340	0.260	0.180
0.425	0.320	0.280	0.220	0.160
0.450	0.280	0.240	0.200	0.140
0.475	0.240	0.210	0.170	0.130
0.500	0.220	0.190	0.155	0.115
0.525	0.195	0.170	0.140	0.100
0.550	0.170	0.150	0.125	0.087
0.575	0.160	0.135	0.112	0.079
0.600	0.140	0.120	0.100	0.069
0.625	0.128	0.110	0.090	0.060
0.650	0.120	0.100	0.080	0.053
0.675	0.108	0.090	0.071	0.048
0.700	0.100	0.085	0.065	0.045
0.725	0.095	0.080	0.061	0.041
0.750	0.090	0.075	0.057	0.040
0.775	0.085	0.070	0.054	0.039
0.800	0.082	0.068	0.052	0.039
0.825	0.080	0.066	0.051	0.038
0.850	0.078	0.065	0.050	0.038

Period Sec.	Damping Ratio 0.005	0.010	0.020	0.050
0.875	0.076	0.064	0.049	0.037
0.900	0.074	0.063	0.049	0.037
0.925	0.073	0.062	0.048	0.037
0.950	0.072	0.062	0.048	0.036
0.975	0.071	0.061	0.047	0.036
1.000	0.070	0.060	0.047	0.036
1.250	0.060	0.051	0.040	0.030
1.500	0.050	0.042	0.033	0.025
1.750	0.040	0.033	0.026	0.020
2.000	0.030	0.025	0.020	0.015

TABLE 7-14

SUMMARY OF RP/RI SEISMIC INTERFACE LOADS

	SSE	OBE
RP Vessel/RI Interface		
CB Flange/Vessel Ledge Load, Vert.	.1147E07	.7647E06
Max. UPS Flange Vessel Head Load, Vert.	.5698E06	.3799E06
Max. Vessel C/B Flange Load, Hor., Lbs.	0.	0.
Max. Vessel/Upper Flange Load, Hor., Lbs.	0.	0.
Max. Upper Core Plate Alignment Pin		
(1) Circumf. Dir. Per pin, lbs.	.1838E05	.1226E05
(2) Radial Dir. Per pin, lbs.	0.	0.
Max. CB/core plate Hor., Lbs.	0.	0.

8.0 SUMMARY

The proposed reactor vessel upper internals replacement described in this Safety Evaluation represents a change to the existing Prairie Island Unit 1 reactor vessel upper internals configuration. As required by 10CFR50.59 a review of this change has been conducted and constitutes the balance of this Safety Evaluation. It has been determined that:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

(ii) The possibility that an accident or malfunction of a different type than any evaluated previously in the safety analysis report does not exist.

(iii) The margin of safety as defined in the basis for any technical specification is not reduced.

This Safety Evaluation provides assurance that Prairie Island Unit 1 can operate safely under all licensed conditions with the reactor vessel upper internals replacement assembly installed.

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