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Susquehanna SES Unit 2
Cycle 3

RELOAD SUMMARY REPORT

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SUSQUEHANNA SES UNIT 2 CYCLE 3
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NOTICE

This technical report was derived in part through information provided to PP&L by Advanced Nuclear Fuels Corporation (formerly Exxon Nuclear Company). It is being submitted by PP&L to the U.S. Nuclear Regulatory Commission specifically in support of the Susquehanna Steam Electric Station Unit 2 Cycle 3 reload. In demonstrating compliance with the U.S. Nuclear Regulatory Commission's regulations, the information contained herein is true and correct to the best of PP&L's knowledge, information, and belief.

CONTENTS

1.0 INTRODUCTION

Susquehanna Steam Electric Station (SSES) Unit 2 Cycle 3 will be the second reload of Advanced Nuclear Fuels Corporation (formerly Exxon Nuclear Company) 9x9 fuel in SSES Unit 2. This report provides a general discussion and summarizes the results of the reload analyses performed by Advanced Nuclear Fuels Corporation (ANF) in support of SSES Unit 2 Cycle 3 (U2C3) operation. Also addressed are a description of the ANF U2C3 reload fuel (XN-2) and core design, and a brief discussion of the license amendment (i.e., proposed Technical Specification changes). The analyses, evaluations, and results presented in this report and the reports referenced herein are similar to those submitted in support of SSES Unit 2 Cycle 2 operation (References 1 and 2) which were approved by the NRC (Reference 3).

The ANF U2C3 Reload Analysis Report ANF-87-126 (Reference 4) and U2C3 Plant Transient Analysis Report ANF-87-125 (Reference 5), along with the proposed changes to the SSES Technical Specifications serve as the basic framework for the reload licensing submittal. When appropriate, reference is made to these and other supporting documents for more detailed information and/or specifics of the applicable analysis. The ANF Reload Analysis Report is intended to be used in conjunction with ANF topical report XN-NF-80-19(P)(A), Vol. 4 Rev. 1, "Application of the ENC Methodology to BWR Reloads" (Reference 6) which describes in more detail the analyses performed in support of the reload and identifies the methodology used for those analyses. The list of references provided at the end of this document contains the SSES specific reload documents prepared by ANF and the applicable ANF generic reload documents (generic methodology previously approved or currently under review) which are being used in support of the U2C3 reload submittal.

The stability issue for ANF 9x9 fuel has been addressed through several calculations (Section 4.2.4 of Reference 4), a startup test performed on SSES Unit 2 (Reference 7), and implementation of Detect and Suppress Technical Specifications.

2.0 GENERAL DESCRIPTION OF RELOAD SCOPE

During the second refueling and inspection outage at SSES Unit 2, PP&L will be replacing 236 fuel assemblies (approximately 31 percent) of the previous Cycle 2 core with fresh XN-2 fuel assemblies. The U2C3 XN-2 fuel is the ANF 9x9 design. The XN-2 fuel has similar operating characteristics (thermal-hydraulic, and nuclear) to the XN-1 design, which has previously been approved (Reference 3) for coresidence with the GE 8x8 fuel that will remain in the core. The Cycle 3 reload core required the performance of a wide range of analyses to support U2C3 operation. These included analyzing for anticipated operational occurrences and for the rapid drop of a high worth control rod to assure that excessive energy would not be deposited in the fuel. In addition, analyses were performed to support Single Loop Operation (SLO) during U2C3. Analyses for normal operation of the reactor consisted of fuel evaluations in the areas of mechanical, thermal-hydraulic, and nuclear design.

Based on ANF's design and safety analyses of the Cycle 3 reload core, a number of proposed changes to the SSES Unit 2 Technical Specifications have resulted. The rationale used to arrive at these proposed changes is contained in the discussions and documentation that follow.

A list of those Technical Specifications and applicable Bases PP&L proposes to change is given below:

Proposed Technical Specification Changes

3/4.2.1 - APLHGR
3/4.2.2 - APRM Setpoints
3/4.2.3 - MCPR Operating Limits
3/4.2.4 - Linear Heat Generation Rate
3/4.4 - Reactor Coolant System

Proposed Changes to Technical Specification Bases

2.1 - Safety Limits

3/4.2 - Power Distribution Limits

3/4.4 - Reactor Coolant System

3.0 SSES UNIT 2 CYCLE 2 OPERATING HISTORY

To date, Cycle 2 has operated with power distributions that will yield end-of-cycle power and exposure shapes consistent with the planned operating strategy. Actual core follow operating data at the time of the reload design analysis was used, together with projected plant operation, as a basis for the Cycle 3 core design and as input to the plant safety analyses. Cycle 2 will be operated within the assumptions of the Cycle 3 analysis; therefore, the remainder of cycle 2 operation will not affect the licensing basis of the Cycle 3 reload core.

4.0 RELOAD CORE DESCRIPTION

The U2C3 core will consist of 764 fuel assemblies, which includes 236 fresh ANF 9x9 assemblies, 324 once burned ANF 9x9 assemblies (XN-1), and 204 initial core GE P8x8R assemblies. The XN-2 reload fuel consists of 140 bundles which contain nine burnable poison rods with 4.0 w/o Gd_2O_3 (9Gd4) and 96 bundles which contain 10 burnable poison rods with 5.0 w/o Gd_2O_3 (10Gd5). A breakdown by bundle type/bundle average enrichment is provided in the following table:

Number of Bundles	Bundle Type	
140	ANF 9x9/3.33 w/o U235	XN-2 (9Gd4)
96	ANF 9x9/3.33 w/o U235	XN-2 (10Gd5)
324	ANF 9x9/3.31 w/o U235	XN-1
196	GE P8x8R/2.19 w/o U235	GE Type III
8	GE P8x8R/1.76 w/o U235	GE Type II

The anticipated Cycle 3 core loading configuration along with additional core design details is provided in Section 4.0 of the ANF U2C3 Reload Analysis Report (Reference 4). The core is essentially a conventional scatter loading with the lowest reactivity bundles placed towards the peripheral region of the core. The loading pattern was designed to maximize the operating cycle length consistent with the constraints on power peaking.

5.0 FUEL MECHANICAL DESIGN

The mechanical design and supporting analyses of the XN-2 fuel are the same as those for the SSES Unit 2 Cycle 2 XN-1 fuel and are described in ANF-87-126 (Reference 4), XN-NF-85-67(P)(A), Revision 1 (Reference 8), XN-NF-84-97 (Reference 9), and PLA-2728 (Reference 10). Each XN-2 reload fuel assembly contains 79 fueled rods and two water rods in a 9x9 rod array. One of the water rods functions as a spacer capture rod. Seven spacers maintain fuel rod spacing.

Generic mechanical design analyses were performed to evaluate the steady state strain, transient strain, cladding fatigue, creep collapse, cladding corrosion, hydrogen absorption, differential fuel rod growth, and grid spacer spring design. These analyses are applicable to the XN-1 and XN-2 fuel designs and support a peak assembly discharge exposure of 40,000 MWD/MTU for ANF 9x9 fuel. The RODEX2, RODEX2A, RAMPEX and COLAPX codes were used in the generic mechanical design analyses. All parameters meet their respective design limits as shown in References 4 and 8. Reference 8 also provides data that demonstrates acceptable rod bow performance for the 9x9 fuel out to 23,000 MWD/MTU. Based on calculations, U2C3 operation is projected to result in a peak XN-1 assembly exposure of greater than 23,000 MWD/MTU but less than 30,000 MWD/MTU at EOC-3. This exposure does exceed the limit specified in Reference 8 but is within the bounds of the exposure limit currently under NRC review (Reference 11). Reference 11 provides additional rod bow measurements on 9x9 Lead Test Assemblies which support an assembly discharge exposure of 40,000 MWD/MTU.

For the initial cycle, GE provided an LHGR design limit to assure operation within the fuel mechanical design analysis, which was incorporated into the Technical Specifications as an operating limit. In addition, a Technical Specification provision for reducing the APRM scram and rod block settings by Fraction of Rated Power divided by Maximum Fraction of Limiting Power Density (FRP/MFLPD) was incorporated to ensure operation within the mechanical design analyses during transients initiated from reduced power. For Cycle 3, this approach will be maintained for the GE fuel.

For the XN-1 and XN-2 ANF 9x9 fuel, the design is such that margin to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) is assured for all anticipated operational occurrences throughout the life of the fuel as demonstrated by the fuel design analysis (Reference 8), provided that the fuel rod power history remains within the power histories assumed in the analysis. This design power profile is shown in Figure 3.3 of Reference 8 and is incorporated into the Technical Specifications as an operating limit. In addition, a Technical Specification provision for reducing the APRM scram and rod block settings by Fraction of Rated Power divided by Maximum Fraction of Limiting Power Density (FRP/MFLPD) was incorporated. This ensures that the ANF 9x9 fuel does not exceed design limits during an overpower condition for transients initiated from partial power. The LHGR curve used for calculating MFLPD for ANF fuel is based on ANF's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of Reference 8 and is incorporated into the Technical Specifications. The Technical Specification curve represents the LHGR corresponding to the ratio of PAFF/1.2, under which cladding and fuel integrity (i.e., 1% clad strain and fuel centerline melting) is protected during AOO's.

The mechanical response of the ANF 9x9 assembly design during Seismic-LOCA events is essentially the same as the response of a GE assembly since the physical properties and bundle natural frequencies are similar as discussed in ANF-87-126 Appendix B (Reference 4). Reference 12 presents the Seismic-LOCA analysis for the GE fuel which shows that resultant loadings do not exceed the fuel design limits. In addition,

Reference 9 presents the Seismic-LOCA analysis for ANF 9x9 fuel in a similar application which showed large design margins for all assembly components. Because the dynamic structural response of the reload core is essentially the same as that of the initial core and large margins are calculated for ANF fuel in a similar application, it is assured that the seismic loads for SSES Unit 2 do not exceed design limits for ANF 9x9 fuel assembly components. Additional justification (Reference 10) was also provided to the NRC by PP&L during the Unit 2 Cycle 2 fuel design review.

6.0 THERMAL HYDRAULIC DESIGN

XN-NF-80-19(P)(A), Volume 4 Revision 1 (Reference 6) presents the primary thermal hydraulic design criteria which require analyses to determine: (1) hydraulic compatibility of the ANF and GE fuel bundles, (2) the fuel cladding integrity safety limit, (3) bypass flow characteristics, and (4) thermal-hydraulic stability. The analyses performed to determine each of these parameters are discussed in this section.

6.1 Hydraulic Compatibility

Component hydraulic resistances for ANF 9x9 fuel and GE P8x8R fuel have been determined in single phase flow tests of full scale assemblies. Reference 1 summarizes the resistances and evaluates the effects on thermal margin due to the coresidence of the ANF 9x9 and GE P8x8R fuel bundles for Unit 2 Cycle 2. Since the XN-2 9x9 design is hydraulically equivalent to the XN-1 9x9 design, this evaluation is also applicable to U2C3 operation. The NRC has previously approved coresidence of GE P8x8R and ANF 9x9 fuel for Unit 2 (Reference 3).

6.2 Safety Limit MCPR

The MCPR fuel cladding integrity safety limit for U2C3 is 1.06 which is equal to the Unit 2 Cycle 2 MCPR safety limit. The methodology and generic uncertainties used in the MCPR safety limit calculation

are provided in XN-NF-80-19(P)(A), Volume 4 Revision 1 (Reference 6). The SSES Unit 2 specific inputs and MCPR Safety Limit calculation are provided in ANF-87-125, Appendix B (Reference 5).

6.3 Core Bypass Flow

Core bypass flow is calculated using the methodology described in XN-NF-524(A) (Reference 13). The core bypass flow fraction (including water rod flow) for U2C3 is 10.1% of total core flow which is similar to the Cycle 2 bypass flow value of 10.3%. The bypass flow fraction is used in the MCPR safety limit calculation and as input to the cycle specific transient analyses.

6.4 Core Stability

COTRAN core stability calculations performed for U2C3 predict stable reactor operation outside of the detect and suppress region of operation in SSES Unit 2. The detect and suppress region is defined by the area above and to the left of the 80% Rod Block line, the 45% constant flow line, and the line connecting the 66% Power/45% Flow, 69% Power/47% Flow points extrapolated to the APRM Rod Block line. Operation outside or on the boundary of this region is supported by COTRAN calculations which result in decay ratios of less than or equal to 0.75 as required by the NRC SER on COTRAN (Reference 14). This region is slightly larger than the region previously specified for SSES Unit 2. The results of this analysis are presented in Reference 4.

PP&L has performed a stability startup test in SSES Unit 2 during initial startup of Cycle 2 to demonstrate stable reactor operation with ANF 9x9 fuel. The test results (Reference 7) show very low decay ratios with a core containing 324 ANF 9x9 fuel assemblies.

SSES Unit 2 Technical Specifications have implemented surveillances for detecting and suppressing power oscillations. This along with the tests and analyses described above assures SSES Unit 2 complies

with General Design Criteria 12, Suppression of Reactor Power Oscillations.

7.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the U2C3 reload are described in the ANF topical report XN-NF-80-19(A), Vol. 1, and Vol. 1 Supplements 1 and 2 (Reference 15). These methods have been reviewed and approved by the Nuclear Regulatory Commission for generic application to BWR reloads.

7.1 Fuel Bundle Nuclear Design

The XN-2 fuel bundle designs are a 9x9 lattice with two (2) inert (water) rods and 79 fuel rods containing 150 inches of active fuel. The top six (6) inches of each fuel rod contain natural uranium and the lower 144 inches (enriched zone) of each rod contain enriched uranium at one of six different enrichments. The XN-2 reload fuel consists of 140 bundles which contain nine burnable poison rods with 4.0 w/o Gd_2O_3(9Gd4) blended with UO_2 enriched to 3.27 w/o U-235 and 96 bundles which contain 10 burnable poison rods with 5.0 w/o Gd_2O_3(10Gd5) blended with UO_2 enriched to 3.27 w/o U-235. These Gd_2O_3 rods are utilized to reduce the initial reactivity of the bundle.

The average enrichment of the enriched zone is 3.44 w/o U235 and the bundle average enrichment (including the top natural uranium blanket) is 3.33 w/o U235. The number of fuel rods at each enrichment is given below:

Enrichment (w/o U235) of Enriched Zone	Number of Rods 9Gd4	10Gd5
1.45	1	1
1.95	5	5
2.55	16	16
3.27	29 (9 contain 4 w/o Gd_2O_3)	29 (10 contain 5 w/o Gd_2O_3)
4.23	13	13
4.66	15	15

The neutronic design parameters and pin enrichment distribution are described in Section 4.0 of the U2C3 Reload Analysis Report (Reference 4).

7.2 Core Reactivity

The beginning-of-cycle 3 (BOC 3) cold core K-effective value with all-rods-out was calculated to be 1.11353. Based on a Cycle 2 length of 10,655 MWD/MTU, a minimum Shutdown Margin of 1.50% ΔK/K, with the strongest worth control rod fully withdrawn at cold (68°F) reactor conditions, was determined to occur at BOC 3. Therefore, R, which is the difference between the minimum Shutdown Margin in the cycle and the BOC Shutdown Margin, is zero. The calculated Shutdown Margin at any point in the cycle is well in excess of the minimum 0.38% ΔK/K Technical Specification requirement, and shall be verified by test at BOC 3 to be greater than or equal to R + 0.38% ΔK/K.

The Standby Liquid Control System, which is designed to inject a quantity of boron that produces a concentration of no less than 660 ppm in the reactor core within approximately 90 to 120 minutes after initiation, was calculated to provide a margin of shutdown of 1.68% ΔK/K with the reactor in a cold, xenon free state, and all control rods at their critical full power positions. This assures that the reactor can be brought from full power to a cold, xenon free shutdown, assuming that none of the withdrawn control rods can be inserted. Thus for the Cycle 3 reload core the basis of the Technical Specification requirement is met.

7.3 Contrast of Cycle 3 Core with Cycle 2

The core loading strategies for Cycle 2 and 3 are very similar in nature. Cycle 2 utilized a conventional scatter loading with the lowest reactivity bundles placed on the core periphery. Cycle 3 will also be based on a scatter loading principle. Fresh reload bundles will be scatter loaded in control cells throughout the core except on the core periphery. The remaining GE Type II bundles will be utilized on the core periphery. Twice burned GE Type III bundles will be utilized on the core periphery and to hold down otherwise "hot" four bundle cells. The once burned XN-1 bundles are distributed throughout the core in a manner which yields acceptable radial peaking, maximizes cycle energy, and provides adequate cold shutdown margin throughout the cycle.

Briefly reviewing the previous fuel bundle designs, the GE Type II and III fuel initially contained axially varying gadolinia at 2, 4, and 5 w/o Gd_2O_3, while the Cycle 2 XN-1 fuel initially contained 4 w/o Gd_2O_3 distributed uniformly over the enriched zones of the designated rods. The Cycle 3 XN-2 fuel bundle designs, in addition to having a higher enrichment, contain two different Gd_2O_3 loadings as described previously in Section 7.1. For reload cycles, the axial exposure profile in the exposed bundles provides an axial shaping effect and eliminates the need for axial gadolinia shaping. Thus, like the XN-1 fuel design, it is not necessary to include axial varying gadolinia in the XN-2 fuel. The XN-2 fuel utilizes an enrichment distribution to yield internal power peaking which results in a balanced and acceptable design relative to MCPR and MAPLHGR Limits. In addition, the XN-1 and XN-2 fuel designs contain a six (6) inch natural uranium section at the top of the fuel bundles in order to increase neutron economy by decreasing leakage at the top of the active core.

7.4 New Fuel Storage Vault/Spent Fuel Pool Criticality

7.4.1 New Fuel Storage Vault

The original neutronics analysis of the currently installed SSES new fuel storage vault was performed by General Electric Company (GE). GE did not limit the stored fuel to a specific enrichment distribution or burnable poison content, but instead limited the $k\infty$ of the fuel lattice (i.e. the maximum enriched zone of the bundle) to $\leq$ 1.30. This insures that, under dry or flooded conditions, the new fuel vault K-Effective remains below 0.95 as specified in the SSES FSAR.

Since the the GE analysis was for an 8x8 lattice, ANF performed calculations for the new fuel vault assuming a 9x9 lattice. The results show that 9x9 fuel with a lattice average enrichment of less than 4 w/o U235 and a $k\infty$ of $\leq$ 1.388 will yield a new fuel vault K-effective $\leq$ .95 under dry or flooded conditions (Reference 16).

The above mentioned $k\infty$ is calculated for a cold (68°F), moderated, uncontrolled fuel assembly lattice in reactor geometry at beginning-of-life (BOL). The maximum cold, uncontrolled, BOL $k\infty$ for the XN-2 fuel assembly enriched zones, as calculated by ANF is 1.10349. This value is well below the ANF analysis criterion of 1.388. Thus for the XN-2 fuel it is concluded that adequate margin to prevent new fuel vault criticality under dry or flooded conditions exists.

Although the new fuel vault has not been designed to preclude criticality at optimum moderation conditions, watertight covers are used, administrative procedures are in place to prevent this condition, and criticality monitors have been installed as an added precaution.

7.4.2 Spent Fuel Pool

The original neutronics analysis for the spent fuel pool as presented in the FSAR was performed by Utility Associates International (UAI). The basis of the analysis assumed the spent fuel pool was loaded with an infinite array of fresh 8x8 fuel assemblies at a uniform average enrichment of 3.25 w/o U235 containing no burnable poison. The absence of burnable poisons insures that peak assembly reactivity occurs at BOL.

ANF performed an analysis to determine criteria for ANF 9x9 fuel that will ensure that the SSES Spent Fuel Pool K-effective will be $\leq$ .95 (Reference 17). The resulting criteria are that the maximum enriched zone of a 9x9 assembly contains less than 4 w/o U235, and that it have an uncontrolled, zero void, 68°F, reactor geometry $k_\infty \leq 1.457$ throughout life. The U2C3 XN-2 fuel design is the ANF 9x9 design and has an enriched zone average enrichment of 3.44 w/o U235. The maximum ANF k_∞ at the appropriate BOL conditions is 1.10349, and at peak reactivity the corresponding k_∞ is 1.2206. These values for the XN-2 fuel designs are significantly lower than the criterion of 1.457 and the enrichment is less than 4.0 w/o U235. Thus, it is concluded that adequate margin exists to prevent spent fuel pool criticality throughout the XN-2 fuel assembly lifetime.

8.0 CORE MONITORING SYSTEM

The POWERPLEX core monitoring system will be utilized to monitor reactor parameters during Cycle 3 and for future ANF reload cycles at SSES. POWERPLEX incorporates ANF's core simulation methodology and is used for both online core monitoring as well as an off-line predictive and backup tool.

The system has been operational at SSES and utilized to monitor reactor parameters during Unit 1 Cycles 2 and 3 and Unit 2 Cycle 2. POWERPLEX is

fully consistent with ANF's nuclear analysis methodology as described in XN-NF-80-19(A) Volume 1 and Volume 1 Supplement 2 (Reference 15). In addition, the measured power distribution uncertainties are incorporated into the calculation of the MCPR Safety Limit as described in ANF's Nuclear Critical Power Methodology Report XN-NF-524(A) (Reference 13).

9.0 ANTICIPATED OPERATIONAL OCCURRENCES

In order to determine operating limits for U2C3 fuel, eight categories of core-wide transients were considered as described in ANF's Plant Transient Methodology Report XN-NF-79-71(P) (Reference 18). ANF has provided analysis results for the following four core-wide transients to determine the thermal margin for U2C3:

1) Generator Load Rejection without Bypass (LRWOB)
2) Feedwater Controller Failure (FWCF)
3) Loss of Feedwater Heater (LOFWH)
4) Recirculation Flow Controller Failure - Increasing Flow

As shown in XN-NF-79-71(P) (Reference 18), the other core-wide transients are non-limiting (i.e., bounded by one of the above). In addition, two local events, Rod Withdrawal Error and Fuel Loading Error, were analyzed in accordance with the methodology described in XN-NF-80-19(A) Vol. 1 (Reference 15). The results of the core-wide and local transient analyses are provided in the U2C3 Reload Analysis Report ANF-87-126 (Reference 4) and in the U2C3 Plant Transient Analysis Report ANF-87-125 (Reference 5). These documents describe the correspondence between the generic documents listed above and the U2C3 specific cases. The core wide transient ΔCPRs are calculated with the XCOBRA-T methodology as described in Reference 19. At rated power and flow conditions the Rod Withdrawal Error, 108% Rod Block Monitor (RBM) setting, was determined to be the limiting event for U2C3 resulting in a ΔCPR of 0.26, and when combined with the 1.06 Safety Limit, requires a MCPR operating limit of 1.32. At less than rated power, the Feedwater Controller failure event is limiting, and at less than rated flow, the Recirculation Flow Control Failure is limiting. Therefore, the MCPR operating limit increases at reduced power and/or flow conditions.

Analyses were also performed to determine the MCPR operating limits with the turbine bypass system inoperable and with the End-of-Cycle Recirculation Pump Trip logic (EOC-RPT) inoperable. The resulting ΔCPRs are 0.28 for the bypass system inoperable and 0.37 for the EOC-RPT inoperable. Therefore, operation with the bypass system inoperable requires a MCPR operating limit of 1.34, and operation with the EOC-RPT inoperable requires a MCPR operating limit of 1.43.

9.1 Core-Wide Transients

The plant transient model used to evaluate the Load Reject without Bypass (LRWOB) and Feedwater Controller Failure (FWCF) event is ANF's COTRANSA code (Reference 18) which incorporates a one-dimensional neutronics model to account for shifts in axial power shape resulting from rapid pressurization and void collapse, and a multi-node steam line model to accommodate pressure waves in the steam line. The ΔCPRs for the LRWOB and FWCF transients are determined by the XCOBRA-T methodology described in Reference 19. All core-wide transients are analyzed deterministically (i.e., using bounding values as input parameters). These analyses cover the turbine bypass system inoperable and End-of-Cycle Recirculation Pump Trip function (EOC-RPT) inoperable conditions. The Loss of Feedwater Heater event (LOFWH) was analyzed deterministically with ANF's PTSBWR code (Reference 18) which uses a point-kinetics neutronics model since rapid pressurization and void collapse do not occur for this event. The LOFWH event yields a less limiting MCPR limit than the LRWOB or FWCF.

Technical Specification scram times were used in the pressurization analyses. Therefore, the calculated operating limit MCPR is conservative for scram times less than the Technical Specification scram times, and no scram speed adjustment to the MCPR operating limit is required for Cycle 3 operation of SSES Unit 2.

9.2 Local Transients

As shown in ANF-87-126 (Reference 4), the results of the Fuel Loading Error are bounded by the Rod Withdrawal Error (RWE) event and is therefore non-limiting. RWE analyses were performed to support an RBM setpoint of 108%. The ΔCPR for the RWE event with a 108% full flow RBM setpoint is 0.26. The RWE event is the limiting event for U2C3 operation and results in a MCPR operating limit of 1.32.

9.3 Reduced Flow/Power Operation

ANF has provided MCPR operating limits for manual flow control reduced flow operation for Cycle 3 in ANF-87-125 (Reference 5). These values are based on ANF's analysis of the recirculation pump flow increase event from reduced flow operation for U2C3. The operating limit consists of a plot of MCPR versus core flow.

The FWCF event is analyzed at reduced power conditions using the XCOBRA-T methodology (Reference 19). A curve of MCPR versus power is included in the Technical Specifications as a power dependent MCPR operating limit.

For all power/flow conditions, the U2C3 MCPR operating limit will be the maximum of the flow dependent operating limit taken from the MCPR versus flow curve and the power dependent operating limit taken from the MCPR versus power curve.

Since the automatic load following capability has been removed from SSES Unit 2, analyses for the automatic flow control mode of operation have not been performed.

9.4 ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code overpressurization criterion of 110% of vessel design pressure, the MSIV closure event with failure of the MSIV position switch scram was analyzed with ANF's COTRANSA code. The U2C3 analysis assumes six safety relief valves are out of service. The maximum pressure observed in the analysis (at the vessel bottom) is 1297 psig or 104% of reactor vessel design pressure, which is within the 110% design criterion.

The calculated steam dome pressure corresponding to the 1297 psig peak vessel pressure is 1281 psig, for a vessel differential pressure of 16 psi. This includes the effects of the ATWS RPT which is assumed to initiate at a pressure setpoint of 1170 psig. The current Technical Specification Safety Limit of 1325 psig is based on dome pressure and therefore conservatively assumes a 50 psi vessel differential pressure (1375-1325). Because the calculated vessel differential pressure is 16 psi, the choice of 1325 psig assures compliance with the ASME criterion of 1375 psig peak vessel pressure while also maintaining consistency with the U2C3 pressure safety limit.

10.0 POSTULATED ACCIDENTS

ANF has previously analyzed the Loss-of-Coolant Accident (LOCA) to determine the MAPLHGR limit for the Unit 2 Cycle 2 XN-1 9x9 fuel. This MAPLHGR limit is also applicable to the U2C3 XN-2 fuel. ANF also performed the Control Rod Drop Accident (CRDA) analysis to demonstrate compliance with the 280 cal/gm Design Limit. The results of these analyses are presented in Section 6.0 of ANF-87-126 (Reference 4). ANF's methodology for the CRDA analysis is described in XN-NF-80-19(A) Vol. 1 (Reference 15) and for the LOCA analysis is provided in References 20 thru 22.

10.1 Loss-of-Coolant Accident

XN-NF-84-117(P) (Reference 23) describes ANF's generic jet pump BWR-4 LOCA break spectrum analysis. This determined the limiting break for BWR-4's with modified Low Pressure Coolant Injection logic to be a double-ended guillotine break in the recirculation piping on the discharge side of the pumps with an assumed discharge coefficient of 0.4 which is equivalent to a total break area of 2.8 ft^2. The analysis of this event for SSES Unit 2 is provided in XN-NF-86-65 (Reference 24). ANF-87-126 (Reference 4) confirms that the MAPLHGR limits in XN-NF-86-65 will ensure that for the U2C3 XN-2 fuel, the peak cladding temperature (PCT) remains below 2200°F, local $Zr\text{-}H_2O$ reaction remains below 17%, and core-wide hydrogen production remains below 1% for the limiting LOCA event as required by 10CFR50. The limiting operating condition was identified in XN-NF-86-65 as the highest power and highest flow permitted by the operating map. The results reported in ANF-87-126 (Reference 4) are bounding for reactor operating conditions up to 100% rated power and 100% rated flow and assure acceptable Peak Cladding Temperatures for the XN-2 fuel during a postulated LOCA event. The LOCA analysis of XN-NF-86-65 (Reference 24) was performed for an entire core of 9x9 fuel and therefore provides MAPLHGR limits for ANF 9x9 fuel only.

The GE LOCA analysis assumed a full core of GE 8x8 fuel; and, as discussed in Sections 6.0 and 7.0, the ANF 9x9 and GE 8x8 fuel are hydraulically and neutronically compatible with respect to core coresidence. Therefore, GE LOCA analysis and MAPLHGR limits for the GE Types II and III fuel remain applicable during U2C3 and for future cycles with mixed GE 8x8/ANF 9x9 cores. The GE Type III MAPLHGR limit curve, for U2C3, contains an additional exposure point at 40,675 MWD/MTU. This point does not appear on the current Unit 2 Cycle 2 MAPLHGR curve. However, this new curve has previously been submitted to and approved by the NRC (Reference 25) for Unit 1 Cycle 3 operation. The MAPLHGR limt is fuel type dependent, therefore, it is applicable to GE Type III fuel in Unit 2 as well as Unit 1.

10.2 Control Rod Drop Accident

ANF's methodology for analyzing the Control Rod Drop Accident (CRDA) is described in XN-NF-80-19(A) Vol. 1 (Reference 15) and utilizes a generic parametric analysis which calculates the fuel enthalpy rise during postulated CRDA's over a wide range of reactor operating variables. The U2C3 analysis was performed consistent with the assumptions used in the U1C4 analysis presented in Reference 26. For U2C3, Section 6.0 of ANF-87-126 (Reference 4) shows a value of 205 cal/gm for the maximum fuel rod enthalpy and less than 250 fuel rods exceeding 170 cal/gm during the worst case postulated CRDA. The 205 cal/gm value is well below the design limit of 280 cal/gm and less than 250 fuel rods exceeding 170 cal/gm is bounded by the 770 rods assumed in Section 15.4.9 of the SSES FSAR (Reference 27). To ensure compliance with the CRDA analysis assumptions, control rod sequencing below 20% core thermal power must comply with GE's Banked Position Withdrawal Sequencing constraints (Reference 28).

11.0 SINGLE LOOP OPERATION

To support Single Loop Operation (SLO) for U2C3, ANF analyzed the MCPR Safety Limit and Recirculation Pump Seizure event considering SLO power/flow conditions and associated SLO uncertainties. The results presented in ANF-87-125, Appendix A (Reference 5) show that the MCPR Safety Limit must increase by 0.01 when in SLO. The 0.01 increase in the Safety Limit is a result of the increased measurement uncertainties associated with SLO. The pump seizure accident is more severe under SLO than under two loop operation, assuming seizure of the operating loop, and is the limiting event over most of the single loop power/flow operating domain. The ΔCPR for the SLO pump seizure event was determined to be 0.30. When operating at low power/flow conditions the two loop events remain limiting.

Previous analyses (References 29 and 30) have shown that other events which could be affected by SLO were non-limiting when analyzed under SLO conditions. For example, the SLO LOCA analysis presented in XN-NF-86-125 (Reference 31) is bounded by the two loop LOCA event.

Based on the vessel internal vibration analysis performed by GE the 80% recirculation pump speed restriction, previously discussed in Reference 29, is maintained for U2C3 SLO.

The results discussed previously in Section 6.4 on core stability also apply under SLO conditions. One of the stability tests performed during the startup of Susquehanna SES Unit 2 Cycle 2, was performed under SLO conditions. The measured decay ratio was 0.30 (σ=0.064) at 55% power/44% flow. ANF performed an analysis of these tests with their COTRAN computer code and calculated a decay ratio of 0.29. This data, the stability calculation results presented in ANF-87-126 (Reference 4), and the S2C3 Technical Specification stability surveillance requirement support SLO during S2C3.

REFERENCES

1. PLA-2661, "Proposed Amendment 39 to License No. NPF-22," B. D. Kenyon (PP&L) to E. Adensam (NRC), June 19, 1986.

2. PLA-2693, "Supplement to Proposed Amendment 39 to License No. NPF-22," B. D. Kenyon (PP&L) to E. Adensam (NRC), July 25, 1986.

3. Letter from E. G. Adensam (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 31 to Facility Operating License No. NPF-22 - Susquehanna Steam Electric Station-Unit 2," October 3, 1986.

4. ANF-87-126, Revision 1, "Susquehanna Unit 2 Cycle 3 Reload Analysis Design and Safety Analyses," Advanced Nuclear Fuels Corporation, November 1987.

5. ANF-87-125, Revision 1, "Susquehanna Unit 2 Cycle 3 Plant Transient Analysis," Advanced Nuclear Fuels Corporation, November 1987.

6. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc., June 1986.

7. XN-NF-86-90, Supplement 1, "Susquehanna Unit 2 Cycle 2 Stability Test Results," Exxon Nuclear Company, Inc., January 1987.

8. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc., September 1986.

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question: Seismic/LOCA Analysis of U2C2 Reload," H. W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. Letter from G. N. Ward (ANF) to G. C. Lainas (NRC), "Additional Information on Rod Bowing," March 11, 1987.

12. PLA-1263, "Response to License condition #14," Letter from N. W. Curtis (PP&L) to A. Schwencer (NRC), August 27, 1982.

13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.

14. XN-NF-691(P)(A) and Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores Sensitivity Analyses and Benchmark Analysis," Exxon Nuclear Company, Inc., August 1984.

15. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., March 1983.

16. XN-NF-86-44, Revision 1, "Criticality Safety Analysis Susquehanna New Fuel Storage Vault with Exxon Nuclear Company, Inc. 9x9 Reload Fuel," Exxon Nuclear Company, Inc., May 1986.

17. XN-NF-86-45, Revision 1, "Criticality Safety Analysis Susquehanna Spent Fuel Storage Pool with Exxon Nuclear Company, Inc. 9x9 Reload Fuel," Exxon Nuclear Company; Inc., May 1986.

18. XN-NF-79-71(P), Revision 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1981.

19. XN-NF-84-105(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Advanced Nuclear Fuels Corporation, February 1987.

20. XN-NF-80-19(A), Volumes. 2, 2A, 2B, and 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., September 1982.

21. XN-NF-CC-33(A), Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Company, Inc., November 1975.

22. XN-NF-82-07(A), Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., November 1982.

23. XN-NF-84-117(P), "Generic LOCA Break spectrum Analysis: BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," Exxon Nuclear Company, Inc., December 1984.

24. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," Exxon Nuclear Company, Inc., May 1986.

25. Letter from W. R. Butler (NRC) to H. W. Keiser (PP&L), "Issuance of Amendment No. 64 to Facility Operating License No. NPF-14 - Susquehanna Steam Electric Station, Unit No. 1," May 7, 1987.

26. PLA-2930, "Corrections to Proposed Amendment 100 to License NPF-14," Pennsylvania Power & Light Company, October 15, 1987.

27. Susquehanna Steam Electric Station, Units 1 and 2, Final Safety Analysis Report.

28. NEDO-21231, "Banked Position Withdrawal Sequence," General Electric Company, January 1977.

29. PLA-2885, "Proposed Amendment 52 to License No. NPF-22," Pennsylvania Power & Light Company, June 30, 1987.

30. PLA-2935, "Additional Information on Proposed Amendment 52 to License No. NPF-22," October 30, 1987.

31. XN-NF-86-125, "Susquehanna LOCA Analysis for Single Loop Operation Analysis," Exxon Nuclear Company, Inc., November 1986.