SUSQUEHANNA SES UNIT 2 CYCLE 2
RELOAD SUMMARY REPORT

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NOTICE

This technical report was derived in part through information provided to PP&L by Exxon Nuclear Company, Inc. 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 2 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 2 will include the first reload of Exxon 9x9 fuel in the Susquehanna Units. This report provides a general scope and summarizes the results of the reload analyses performed by Exxon Nuclear Company (ENC) in support of SSES Unit 2 Cycle 2 (U2C2) operation. Also addressed is a description of the ENC U2C2 reload fuel (XN-1) and core design, General Electric (GE) and ENC reload fuel compatibility, and a brief discussion of the license amendment (proposed Tech. Spec. 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 1 Cycle 3 operation (Reference 1) which were approved by the NRC (Reference 2).

The ENC U2C2 Reload Analysis Report XN-NF-86-60 (Reference 3), Susquehanna LOCA-ECCS Analysis XN-NF-86-65 (Reference 4) and U2C2 Plant Transient Analysis Report XN-NF-86-55 (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. A supplemental MCPR report which revises the MCPR limits in the above reports will be provided later. The revised MCPR limits will be calculated with the XCOBRA-T methodology as described in XN-NF-84-105 (Reference 23). The ENC Reload Analysis Report is intended to be used in conjunction with ENC topical report XN-NF-80-19(P), 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 contain the SSES specific reload documents prepared by ENC and the applicable ENC Generic Reload Documents (generic methodology previously approved or currently under review) which are being used in support of the U2C2 reload submittal. References 3 and 5 contain analyses for operation with Increased Core Flow and Final Feedwater Temperature Reduction. Some of the operating limits incorporated into the Technical Specifications are applicable for the Increased Core Flow region of operation. However, PP&L is not supporting U2C2 operation with either option at this time, and the Technical Specifications preclude operation above rated core flow.

The stability issue for ENC 9x9 fuel has been addressed through several calculations (Section 6.4 and Reference 3), a startup test plan (Reference 7), and implementation of Detect and Suppress Technical Specifications.

2.0 GENERAL DESCRIPTION OF RELOAD SCOPE

During the first refueling and inspection outage at SSES Unit 2, PP&L will be replacing 324 fuel assemblies (approximately 42 percent) of the previous Cycle 1 core with fresh XN-1 fuel assemblies. The U2C2 XN-1 fuel is the ENC 9x9 design, which has similar operating characteristics (thermal-hydraulic, and nuclear) to the GE P8x8R fuel that will remain in the core. The differences in mechanical and nuclear design required a wide range of analyses to be performed. These included analyzing for anticipated operational occurrences, performing LOCA and MAPLHGR analyses for the XN-1 fuel, and analyzing for the rapid drop of a high worth control rod to assure that excessive energy would not be deposited in the fuel. Analyses for normal operation of the reactor consisted of fuel evaluations in the areas of mechanical, thermal-hydraulic, and nuclear design. In addition, changes were also implemented to the core monitoring system and supplemental analyses were performed to reevaluate the expanded power flow map region for Cycle 2 operation.

Based on ENC's design and safety analyses of the Cycle 2 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.

Although Exxon provides a discussion in Appendix A of XN-NF-86-60 (Reference 3), PP&L is not supporting the use of Single Loop Operation

(SLO) for U2C2 until SLO (reduced power, flow) LOCA-ECCS MAPLHGR calculations for ENC 9x9 fuel are completed. These will be presented in a separate submittal to the NRC. Therefore, the Unit 2 Technical Specifications have been revised to set the SLO MAPLHGR values to zero, thereby precluding extended operation with one loop out of service until the final analyses are approved.

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

<u>Proposed Technical Specification Changes</u>

1.0 - Definitions
3/4.1.2 - Reactivity Anomalies
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
3/4.3.4.2 - End-of-Cycle Recirculation Pump Trip System
3/4.4 - Reactor Coolant System
3/4.7 - Plant Systems
5.3 - Design Features

<u>Proposed Changes to Technical Specification Bases</u>

2.1 - Safety Limits
3/4.1 - Reactivity Control Systems
3/4.2 - Power Distribution Limits
3/4.4 - Reactor Coolant System
3/4.7 - Plant Systems

3.0 SSES UNIT 2 CYCLE 1 OPERATING HISTORY

To date, Cycle 1 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 2 core design and as input to the plant safety analyses. Cycle 1 has continued to operate as expected and no operating

anomalies have occurred which would significantly affect the licensing basis of the reload core or Cycle 2 performance.

The current end-of-cycle 1 (EOC 1) licensing exposure window ranges from 11,220 MWD/MTU to 13,000 MWD/MTU. This window provides an allowable EOC 1 core average exposure range for which the Cycle 2 plant safety analyses are valid.

4.0 RELOAD CORE DESCRIPTION

The U2C2 core will consist of 764 fuel assemblies, which include 324 fresh XN-1 assemblies, and 440 once burned GE P8x8R assemblies. A breakdown by bundle type/bundle average enrichment is provided in the following table:

Number of Bundles	Bundle Type	
324	ENC 9x9/3.31 w/o U235	XN-1
432	GE P8x8R/2.19 w/o U235	GE Type III
8	GE P8x8R/1.76 w/o U235	GE Type II

Of the 324 once burned GE P8x8R fuel assemblies being discharged at EOC 1, 232 are medium enriched (1.76 w/o U235) bundles, and 92 are natural uranium (.711 w/o U235) bundles which are currently located on the core periphery.

The anticipated Cycle 2 core loading configuration along with additional core design details is provided in section 4.0 of the ENC U2C2 Reload Analysis Report (Reference 3). The core is essentially a conventional scatter load with the lowest reactivity bundles placed in the periphery 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-1 fuel are described in, XN-NF-86-60, XN-NF-85-67 Revision 1 and XN-NF-84-97

Revision 0 (References 3, 8 and 9). Each XN-1 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. The fuel rods are pre-pressurized, contain UO_2 pellets, and are basically of the same design as used in the ENC 8x8 BWR design, except the diameters of the pellets and cladding are smaller. The pellet cladding diametral gap is also slightly smaller than in ENC 8x8 fuel.

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 fuel and support an assembly discharge exposure of 40,000 MWD/MTU. As addressed in XN-NF-85-67 (Reference 8), the rod bow criteria are only supported for two cycles of operation, because the 9x9 Lead Test Assemblies (LTAs) have only been irradiated for two cycles. As the LTAs irradiation increases this lifetime limit will be extended. RODEX2, RODEX2A, RAMPEX and COLAPX codes were used in the generic mechanical design analyses. Cycle specific analyses of fuel rod maximum internal pressure and fuel temperature were performed using RODEX2A. All parameters meet their respective design limits as shown in References 3 and 8.

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 2, this approach will be maintained for the GE fuel.

For XN-1 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 history 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 ENC fuel does not exceed design limits during an overpower condition for transients initiated from partial power. The LHGR curve used for calculating MFLPD for ENC fuel is based on ENC's Protection Against Fuel Failure (PAFF) line shown in Figure 3.4 of Reference 8 and is incorporated in the Technical Specifications. This 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 center line melting) is protected during AOO's.

The mechanical response of the ENC 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 XN-NF-86-60 Appendix B (Reference 3). Reference 10 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 ENC 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 ENC fuel in a similar application, it is assured that the seismic loads for SSES Unit 2 do not exceed design limits for ENC 9x9 fuel assembly components.

6.0 THERMAL HYDRAULIC DESIGN

XN-NF-80-19, Volume 4 (Reference 6) presents the primary thermal hydraulic design criteria which require analyses to determine: (1) hydraulic compatibility of the ENC 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 the ENC fuel and the GE P8x8R fuel have been determined in single phase flow tests of full scale assemblies. XN-NF-86-60 summarizes the resistances and evaluates the effects on thermal margin due to the coresidence of the ENC and GE fuel bundles. The similarity of the performance characteristics of the fuel designs prove that they are compatible for coresidence in SSES Unit 2.

6.2 Safety Limit MCPR

The MCPR fuel cladding integrity safety limit for U2C2 is 1.06 which is equal to the Unit 2 Cycle 1 MCPR safety limit. The methodology and generic uncertainties used in the MCPR safety limit calculation are provided in XN-NF-80-19, Volume 4 (Reference 6). The SSES Unit 2 specific inputs and MCPR Safety Limit calculation are provided in XN-NF-86-55 (Reference 5).

6.3 Core Bypass Flow

Core bypass flow is calculated using the methodology described in XN-NF-524(A) (Reference 12). The core bypass flow fraction for U2C2 is 10.3% of total core flow which is equal to the initial core value. The bypass flow fraction is used in the MCPR safety limit calculation and as input to the transient analysis.

6.4 Core Stability

COTRAN core stability calculations performed for U2C2 predict stable reactor operation outside of the detect and suppress region of operation in Susquehanna Unit 2. The results of this analysis are presented in Reference 3.

PP&L has committed to additional efforts designed to demonstrate stable reactor operation with ENC 9x9 fuel. A stability startup test is planned for U2C2, with post test analysis to verify core stability. This plan is outlined in Reference 7. Per NRC request, core stability calculations for U2C2 will be performed with COTRAN and COTRANSA2 at several points on the power flow map to supplement the analysis in Reference 3. Submittal of these additional stability calculations to the NRC is scheduled for July 1986.

SSES Unit 2 Technical Specifications have implemented surveillances for detecting and suppressing power oscillations. This along with the testing and analyses described above assures SSES U2 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 U2C2 reload are described in the ENC topical report XN-NF-80-19(A), Vol. 1, and Vol. 1 Supplements 1 and 2 (Reference 13). 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-1 fuel bundle design is an 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 fuel bundle burnable poison design consists of seven (7) gadolinia-bearing rods containing 4.0 w/o Gd_2O_3. These rods are utilized to reduce the initial reactivity of the bundle.

The average enrichment of the enriched zone is 3.42 w/o U235 and the bundle average enrichment (including the top natural uranium blanket) is 3.31 w/o U235. The number of fuel rods at each enrichment is given below:

Number of Rods	Enrichment (w/o U235) of Enriched Zone
1	1.45
5	1.95
18	2.58
27	3.27 (7 contain 4.0 w/o GD_2O_3)
13	4.18
15	4.68

The neutronic design parameters and pin enrichment distribution are described in section 4.0 of the U2C2 Reload Analysis Report (Reference 3). A brief summary of the fuel assembly reactivity (k_∞) for the 3.42 w/o U235 enriched zone at various beginning-of-life (BOL) conditions is provided in the following table:

Condition	Fuel Temperature (°F)	Moderator Temperature (°F)	Void Fraction (%)	Uncontrolled k_∞	Controlled k_∞
Cold	68	68	0	1.1501	0.9827
Intermediate	180	180	0	1.1441	0.9703
Hot Standby	548.8	548.8	0	1.1225	0.9105
Hot Operating	892	548.8	0	1.1191	0.9078
Hot Operating	892	548.8	40	1.0994	0.8500
Hot Operating	892	548.8	70	1.0827	0.7971

7.2 Core Reactivity

The beginning-of-cycle 2 (BOC 2) cold core Keff value with all-rods-out was calculated to be 1.1081. Based on the nominal Cycle 1 length of 12,220 MWD/MTU, a minimum Shutdown Margin of 2.71% $\Delta K/K$, with the strongest worth control rod fully withdrawn at cold (68°F) reactor conditions, was determined to occur at a Cycle 2 exposure of 8,000 MWD/MTU. The BOC 2 Shutdown Margin was calculated to be 2.75% $\Delta K/K$. Therefore, the difference between the minimum Shutdown Margin in the cycle and the BOC Shutdown Margin, R, is 0.04% $\Delta K/K$. The calculated Shutdown Margin at any point in the cycle is well in excess of the minimum 0.38% $\Delta K/K$ Technical Specification requirement, and shall be verified by test at BOC 2 to be greater than or equal to R + 0.38% $\Delta K/K$.

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 2.42% $\Delta K/K$ 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 2 reload core the basis of the Technical Specification requirement is met.

7.3 Contrast of Cycle 2 Core with Cycle 1

The differences between the Cycle 1 core and the Cycle 2 core are apparent in both the core loading pattern and fuel bundle design. Cycle 1 is a standard GE BWR/4 initial core configuration consisting of a center "dead cross" region of medium enriched (1.76 w/o U235) bundles, an internal "checker board" region of high (2.19 w/o U235) and medium enriched bundles, a "ring of fire" zone of high enriched bundles, and a zone of natural enriched (.711 w/o U235) bundles located on the core periphery. In contrast, the Cycle 2 core will be based on the conventional scatter load principle. Fresh reload bundles are scatter loaded in control cells throughout the core except on the periphery.

The GE initial core fuel contains axially varying gadolinia at 2, 4, and 5 w/o GD_2O_3 in the enriched zones of designated rods. The XN-1 fuel bundle design, in addition to having a higher enrichment, contains 4 w/o Gd_2O_3 distributed uniformly over the enriched length of the designated rods. This increased gadolinia concentration is necessary in order to maintain adequate reactivity control over a long reload cycle. 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, it is not necessary to include axial varying gadolinia in the XN-1 fuel bundle design, as it was in the GE initial core fuel bundle designs. The XN-1 fuel bundle average enrichment of 3.31 w/o U235 was chosen to provide for the anticipated Cycle 2 energy requirements. This enrichment is higher than the remaining GE fuel as shown in the table in section 4.0. In addition to the enrichment and burnable poison differences, the GE high and medium enriched fuel contains six (6) inch natural uranium sections at both the top and bottom of the fuel bundle, while the XN-1 fuel contains a six (6) inch natural uranium section at the top of the fuel bundle only. Elimination of the bottom natural uranium blanket in the XN-1 fuel bundles results in a negligible increase in neutron leakage at the core inlet while at the same time adding significant improvement to core scram reactivity.

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_∞ of the fuel lattice (i.e. the maximum enriched zone of the bundle) to ≤ 1.30, under dry or flooded conditions, which insures that the new fuel vault K_{eff} remains below 0.95 as specified in the SSES FSAR.

However, since the XN-1 fuel is the ENC 9x9 design, and the GE analysis was for an 8x8 lattice, ENC performed calculations for the new fuel vault. These show 9x9 fuel with a lattice average enrichment less than 4 w/o U235 and a k_∞ of $\leq$ 1.388 will yield a new fuel vault $k_{eff} \leq .95$ under dry or flooded conditions (Reference 4).

The above mentioned k_∞ is calculated for a cold (68^{o}F), moderated, uncontrolled fuel assembly lattice in reactor geometry at beginning-of-life (BOL). The cold, uncontrolled, BOL k_∞ for the XN-1 fuel assembly enriched zone, as calculated by ENC and listed in section 7.1, is 1.1501. This value is well below both the GE analysis criteria of 1.30 and the ENC analysis criteria of 1.388. Thus for the XN-1 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 and 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.

Because the U2C2 XN-1 fuel design is the ENC 9x9 design and has an enriched zone average enrichment of 3.42 w/o U235 the previous analysis is not applicable. ENC performed an analysis to determine criteria for ENC 9x9 fuel that will ensure that the SSES Spent Fuel Pool $K_{effective}$ will be $\leq$ .95 (Reference 15). The resulting criteria are that the 9x9 assembly contain less than 4% U235 on average, and have an uncontrolled, zero void, 68^{o}F, reactor geometry $k^{\infty} \leq 1.457$ throughout life. As stated in section 7.1 the ENC k_{∞} at the appropriate BOL conditions is 1.1501 and at peak reactivity the appropriate k_{∞} is 1.2221. These values for the XN-1 fuel are significantly lower than the criteria and thus it is concluded that adequate margin exists to prevent spent fuel pool criticality throughout the XN-1 fuel assembly lifetime.

8.0 CORE MONITORING SYSTEM

The POWERPLEX core monitoring system will be utilized to monitor reactor parameters during Cycle 2 and for future ENC reload cycles at SSES. POWERPLEX incorporates ENC'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. POWERPLEX is fully consistent with ENC's nuclear analysis methodology as described in XN-NF-80-19(A) Volume 1 and Volume 1 Supplement 2 (Reference 13). In addition, the measured power distribution uncertainties are incorporated into the calculation of the MCPR Safety Limit as described in ENC's Nuclear Critical Power Methodology Report XN-NF-524(A) (Reference 12).

9.0 ANTICIPATED OPERATIONAL OCCURRENCES

In order to determine operating limits for U2C2 fuel, eight categories of core-wide transients are considered as described in ENC's Plant Transient Methodology Report XN-NF-79-71(P) (Reference 16). ENC has provided analysis results for the following two core-wide transients to determine the thermal margin for U2C2:

1) Generator Load Rejection without Bypass (LRWOB)
2) Feedwater Controller Failure (FWCF).

As shown in XN-NF-79-71(P) (Reference 16) and previous Susquehanna Licensing submittals, 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 13). The results of the core-wide and local transient analyses are provided in the U2C2 Reload Analysis Report XN-NF-86-60 (Reference 3) and in the U2C2 Plant Transient Analysis Report XN-NF-86-55 (Reference 5). These documents describe the correspondence between the generic documents listed above and the U2C2 specific cases. A supplemental MCPR report which revises the ΔCPRs resulting from the LRWOB and FWCF transients in the above reports will be provided later. The revised ΔCPRs will be calculated with the XCOBRA-T methodology as described in Reference 23. Because PP&L is not supporting operation in U2C2 with ICF/FFWTR these analyses will not bound those conditions. The Rod Withdrawal Error (106% Rod Block Monitor-RBM setting) resulted in a ΔCPR of 0.17, and when combined with the 1.06 Safety Limit, requires a MCPR operating limit of 1.23. An additional Rod Withdrawal Error analysis was performed to determine the MCPR operating limit with a 108% RBM setpoint. The resulting ΔCPR is 0.21 for a 108% RBM setpoint. Therefore, operation with a 108% RBM setpoint requires a MCPR operating limit of 1.27. The above limit is applicable to cover a flow window from 100% power/87% flow to 100% power/100% flow conditions. Note, the supplemental MCPR report may provide ΔCPRs which result in higher operating limits than are presented herein for the Rod Withdrawal Error.

9.1 Core-Wide Transients

The plant transient model used to evaluate the Load Reject without Bypass (LRWOB) and Feedwater Controller Failure (FWCF) events is ENC's COTRANSA code (XN-NF-79-71, Rev.2) 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 will be determined by the XCOBRA-T methodology described in Reference 23. All core wide transients will be analyzed deterministically (i.e., using bounding values as input parameters). These analyses will cover the turbine bypass system inoperable and End of Cycle recirculation pump trip function (EOC-RPT) inoperable conditions. As shown in previous Susquehanna licensing submittals (e.g., Reference 1), the Loss of Feedwater Heater (LOFWH) event yields a less limiting MCPR limit than the LRWOB or FWCF.

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

9.2 Local Transients

As shown in XN-NF-86-60 (Reference 3), the result of the Fuel Loading Error is bounded by the Rod Withdrawal Error (RWE) event and is therefore non-limiting. Based on the RWE results, the MCPR operating limit is a function of the RBM setpoint. Analyses were performed to support RBM setpoints of 106% and 108% to provide additional flexibility in utilizing the allowable power/flow operating region above the 100% flow control line. The ΔCPR for the RWE event with a 108% full flow RBM setpoint is 0.21, and for a 106% full flow RBM setpoint the ΔCPR is 0.17.

9.3 Reduced Flow/Power Operation

ENC has provided MCPR operating limits for manual flow control reduced flow operation for Cycle 2 in XN-NF-86-55 (Reference 5). These values are based on ENC's analysis of the recirculation pump flow increase event from reduced flow operation for U2C2. The operating limit consists of a plot of MCPR versus core flow.

The FWCF event will be analyzed at reduced power conditions using the XCOBRA-T methodology (Reference 23). It is expected that reduced power MCPR limits will be required. A curve of MCPR versus power will be included in the Technical Specifications as a power dependent MCPR operating limit.

For all power/flow conditions, the U2C2 operating MCPR limit will be the maximum of the flow dependent operating limit taken from the MCPR versus flow curve, the power dependent operating limit taken from the MCPR versus power curve, and the full flow operating limit.

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 criteria of 110% of vessel design pressure, the MSIV closure event with failure of the MSIV position switch scram was analyzed with ENC's COTRANSA code. The U2C2 analysis assumes six safety relief valves are out of service. The maximum pressure observed in the analysis (at the vessel bottom) is 1315 psig or 105% of reactor vessel design pressure, which is within the 110% design criterion.

The calculated steam dome pressure corresponding to the 1315 psig peak vessel pressure is 1301 psig, for a vessel differential pressure of 14 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 14 psi, the choice of 1325 psig assures compliance with the ASME criterion of 1375 psig peak vessel pressure while also maintaining consistency with the U2C2 pressure safety limit.

10.0 POSTULATED ACCIDENTS

In support of U2C2 operation, ENC has analyzed the Loss-of-Coolant Accident (LOCA) to determine MAPLHGR limits for XN-1 fuel, and the Rod Drop Accident (RDA) to demonstrate compliance with the 280 cal/gm Design Limit. The results of these analyses are presented in section 6.0 of XN-NF-86-60 (Reference 3). ENC's methodology for the RDA analysis is described in XN-NF-80-19(A) Vol. 4 (Reference 6) and for the LOCA analysis is provided in References 17 thru 19.

10.1 Loss-of-Coolant Accident

XN-NF-84-117(P) (Reference 20) describes ENC's generic jet pump BWR4 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 pump 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 initiated from 100% power and 108% flow for SSES Unit 2 is provided in XN-NF-86-65 (Reference 4) and summarized in XN-NF-86-60 (Reference 3). Operation within the MAPLHGR limits of Section 6.0 of XN-NF-86-60 (Reference 3) for XN-1 fuel will ensure that the peak cladding temperature (PCT) remains below

$2200^{o}F$, local $Zr-H_2O$ 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 (Reference 4) as the highest power and highest flow permitted by the operating map, which is bounded by the 100% rated power and 108% rated flow condition used in the analysis. Operation in the Increased Core Flow region is precluded by the Technical Specifications, and the results reported in XN-NF-86-60 (Reference 3) are bounding for reactor operating conditions up to 100% rated power and 100% rated flow, and assure acceptable XN-1 Peak Cladding Temperatures (PCT) during a postulated LOCA event. The LOCA analysis of XN-NF-86-65 (Reference 4) was performed for an entire core of XN-1 fuel and therefore provide MAPLHGR's for ENC 9x9 fuel only.

As discussed in Sections 5.0, 6.0, and 7.0 ENC fuel is hydraulically and neutronically compatible with GE fuel. Therefore, the existing GE LOCA Analysis (which is described in the SSES FSAR for Units 1 and 2 and Reference 21 for the expanded power/flow region) and MAPLHGR limits will remain applicable for GE fuel during U2C2 and future cycles with GE/ENC mixed cores.

10.2 Control Rod Drop Accident

ENC's methodology for analyzing the Control Rod Drop Accident (RDA) is described in XN-NF-80-19(A) Vol. 4 (Reference 6) and utilizes a generic parametric analysis which calculates the fuel enthalpy rise during postulated RDA's over a wide range of reactor operating variables. For U2C2, Section 6.0 of XN-NF-86-60 (Reference 3) shows a value of 147 cal/gm for the maximum fuel rod enthalpy during the worst case postulated RDA. This value is well below the design limit of 280 cal/gm. To ensure compliance with the RDA analysis assumptions, control rod sequencing below 20% core thermal power must comply with GE's Banked Position Withdrawal Sequencing constraints (Reference 22).

REFERENCES

1. PLA-2585, "Proposed Amendment 78 to License No. NPF-14," H. W. Keiser to E. Adensam, January 16, 1986.
2. Letter from E. G. Adensam to H. W. Keiser, "Amendment No. 57 to Facility Operating License No. NPF-14, Susquehanna Steam Electric Station," dated April 11, 1986.
3. XN-NF-86-60, "Susquehanna Unit 2 Cycle 2 Reload Analysis," Exxon Nuclear Co., May 1986.
4. XN-NF-86-65, "Susquehanna LOCA-ECCS Analysis MAPLHGR Results for 9x9 Fuel," Exxon Nuclear Co., May 1986.
5. XN-NF-86-55, "Susquehanna Unit 2 Cycle 2 Plant Transient Analysis," Exxon Nuclear Co., May 1986.
6. XN-NF-80-19(P), Vol. 4, Rev. 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Co., March 1985.
7. PLA-2637, "Unit 2 Cycle 2 Stability Test Program," H. W. Keiser to E. Adensam, April 30, 1986.
8. XN-NF-85-67, Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Co., April 1986.
9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Co., December 1984.
10. PLA-1263, "Response To License Condition #14," Pennsylvania Power & Light Co., August 27, 1982.
11. Not used.
12. XN-NF-524(A), Rev. 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Co., November 1983.
13. XN-NF-80-19(A), Vol. 1, and Vol. 1 Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Co., March 1983.
14. XN-NF-86-44, Revision 1, "Criticality Safety Analysis Susquehanna New Fuel Storage Vault with Exxon Nuclear Company, Inc. 9x9 Reload Fuel," Exxon Nuclear Co., May 1986.
15. XN-NF-86-45, Revision 1, "Criticality Safety Analysis Susquehanna Spent Fuel Storage Pool with Exxon Nuclear Company, Inc. 9x9 Reload Fuel," Exxon Nuclear Co., May 1986.

16. XN-NF-79-71(P), Rev. 2, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," Exxon Nuclear Co., November 1981.
17. XN-NF-80-19(A), Vols. 2, 2A, 2B, & 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Co., September 1982.
18. XN-NF-CC-33(A), Rev. 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Co., November 1975.
19. XN-NF-82-07(A), Rev. 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Co., November 1982.
20. XN-NF-84-117(P), "Generic LOCA Break Spectrum Analysis: BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," Exxon Nuclear Co., December 1984.
21. PLA-1506, "Proposed Amendment 19 to License NPF-14," Pennsylvania Power & Light Co., February 3, 1983.
22. NEDO-21231, "Banked Position Withdrawal Sequence," General Electric Co., January 1977.
23. XN-NF-84-105, Volume 1 and Revision 1 of Supplements 1 and 2, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Co., May 1985 and March 1986.