SUSQUEHANNA SES UNIT 1 CYCLE 3
RELOAD SUMMARY REPORT

Prepared by: R. M. Rose
R. M. Rose/Nuclear Fuel Management
Project Engineer

A. J. Roscioli
A. J. Roscioli/Nuclear Systems
Project Engineer

Concur with: J M Kulick
J. M. Kulick/Supervisor -
Core Management

W. J. Rhoades/Supervisor -
Engineering Analysis

Approved by: J. S. Stefanko
J. S. Stefanko/Manager -
Nuclear Fuels & Systems Engineering

PENNSYLVANIA POWER & LIGHT COMPANY

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 1 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.

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1.0 INTRODUCTION

Susquehanna Steam Electric Station (SSES) Unit 1 Cycle 3 will include the second reload of Exxon 8x8 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 1 Cycle 3 (U1C3) operation. Also addressed is a description of the ENC U1C3 reload fuel (XN-2) and core design, GE and ENC reload fuel compatibility, and a brief discussion of the license amendment (proposed Tech. Spec. changes). The analyses and evaluations presented in this report and the reports referenced herein are similar to those submitted in support of SSES Unit 1 Cycle 2 operation (References 1 and 2) which were approved by the NRC in Reference 3.

The ENC U1C3 Reload Analysis Report XN-NF-85-132 (Reference 4) and U1C3 Plant Transient Analysis Report XN-NF-85-130 (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 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 U1C3 reload submittal.

2.0 GENERAL DESCRIPTION OF RELOAD SCOPE

During the second refueling and inspection outage at SSES Unit 1, PP&L will be replacing approximately two fifths of the previous Cycle 2 core with fresh XN-2 fuel assemblies. The XN-2 fuel is very similar to the XN-1 fuel with the exception of small differences in the enrichment and slight differences in the mechanical design. These small differences coupled with the modified core design for Cycle 3 warranted a number of reanalyses to be performed by ENC. This included reanalyzing for anticipated operational occurrences, performing LOCA and MAPLHGR analyses for the XN-2 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.

Based on ENC's design and safety analyses of the Cycle 3 reload core, a number of proposed changes to the SSES Unit 1 Technical Specifications have resulted. The rationale used to arrive at these proposed changes is inherent in the discussions and documentation that follows.

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

Proposed Technical Specification Changes

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

Proposed Changes to Technical Specification Bases

3/4.1 - Reactivity Control Systems
3/4.2 - Power Distribution Limits

3.0 SSES UNIT 1 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 a standard Haling 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 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 3 performance.

The current end-of-cycle 2 (EOC 2) licensing exposure window ranges from 13,850 MWD/MTU to 15,250 MWD/MTU. This window provides an allowable EOC 2 core average exposure range for which the Cycle 3 plant safety analyses are valid.

4.0 RELOAD CORE DESCRIPTION

The U1C3 core will consist of 764 fuel assemblies, which includes 296 fresh XN-2 assemblies, 192 once burned XN-1 assemblies, and 276 twice burned GE P8x8R assemblies. A breakdown by bundle type/bundle average enrichment is provided in the following table:

Number of Bundles	Bundle Type	
296	ENC 8x8/2.89 w/o U235	XN-2
192	ENC 8x8/2.72 w/o U235	XN-1
276	GE P8x8R/2.19 w/o U235	GE Type III

Of the 296 twice burned GE P8x8R fuel assemblies being discharged at EOC 2, 140 are medium enriched (1.76 w/o U235) bundles, 92 of which are currently located on the core periphery, and 156 are high enriched (2.19 w/o U235) bundles.

The anticipated Cycle 3 core loading configuration along with additional core design details is provided in section 4.0 of the ENC U1C3 Reload Analysis Report (Reference 4). 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.

U1C3 is estimated to provide 1431 GWD of energy based on a Cycle 2 energy output of 694.5 GWD.

5.0 FUEL MECHANICAL DESIGN

The mechanical design and supporting analyses of the XN-2 fuel are described in XN-NF-81-21, Revision 1 and Revision 1 Supplement 1 (References 7 and 8). The reload fuel assembly mechanical design is very similar to the XN-1 fuel design. Both fuel types contain 62 fuel rods and two centrally located water rods, one of which functions as a spacer capture rod. Seven spacers maintain fuel rod spacing. The fuel rods are pre-pressurized, contain UO_2 pellets, and use a diametral pellet-to-clad gap which is smaller on the interior high enrichment rods than on the remaining rods in the bundle to improve ECCS margin.

Generic mechanical design analyses were performed to evaluate cladding steady-state strain, transient strain, fatigue damage, creep collapse, corrosion buildup and hydrogen absorption, fuel rod maximum internal pressure, differential fuel rod growth, creep bow, and grid spacer spring design. These analyses are applicable to the XN-2 fuel and support a batch average burnup of 30,000 MWD/MTU. All parameters meet their respective design limits as shown in References 7 and 8. XN-NF-85-132 (Reference 4) presents the fuel thermal analysis that shows no fuel centerline melting at 120% overpower conditions for all exposures within the design end-of-life exposure.

The analyses provided in Reference 7 are approved by the NRC for generic application to ENC fuel with the exception of the design strain, external corrosion, rod pressure, overheating of the fuel pellets, and pellet cladding interaction analyses which were performed using an unapproved version of the RODEX2 fuel performance computer code. A revised version of RODEX2 has recently been approved by the NRC (Reference 9). Reference 8 presents the revised mechanical design analyses using the approved version of RODEX2. These revised analyses were approved for the XN-1 reload of SSES Unit 1 in Reference 3.

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 with excessive peaking (i.e., peaking which would result in a LHGR in excess of the operating limit if power were increased to rated). For Cycle 3, this approach will be maintained for the GE fuel.

For ENC fuel, including XN-2, the design is such that margin to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) is assured for 120% overpower conditions throughout the life of the fuel as demonstrated by the fuel design analyses (Reference 7 and 8) and the fuel thermal analysis (Reference 4), provided that the fuel rod power history remains within the power history assumed in the analysis. This design power profile is shown in Figure 5.10 of Reference 7 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 a 120% overpower condition for transients initiated from partial power with excessive peaking (i.e, peaking which would result in a LHGR in excess of the operating limit if power were increased to rated).

The mechanical response of the ENC 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. Reference 10 presents the Seismic-LOCA analysis for the GE fuel which shows that resultant loadings do not exceed the fuel design limits. Reference 11 presents the Seismic-LOCA analysis for ENC fuel in a similar application which showed large design margins for all assembly components.

Therefore, based on the similarity between the fuel types and the large margin calculated for ENC fuel in a similar application, the seismic loads for SSES Unit 1 do not exceed design limits for ENC 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 8x8 fuel have been determined in single phase flow tests of full scale assemblies. XN-NF-80-19, Volume 4 summarizes the resistances and evaluates the effects on thermal margin due to the coresidence of the ENC and GE fuel bundles. The close similarity between the two fuel designs and their performance characteristics indicate that they are sufficiently compatible for coresidence in SSES Unit 1.

6.2 Safety Limit MCPR

The MCPR fuel cladding integrity safety limit for U1C3 is 1.06 which is equal to the Unit 1 Cycle 2 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 1 specific inputs and MCPR safty limit calculation are provided in XN-NF-85-130 (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 U1C3 is 10.0% of total core flow which is similar to the Cycle 2 core value of 10.4%. The bypass flow fraction is used in the MCPR safety limit calculation and as input to the transient analysis.

6.4 Core Stability

SSES Unit 1 Technical Specifications have implemented surveillances for detecting and suppressing power oscillations. Therefore, as discussed in a letter from PP&L to Mr. A. Schwencer (NRC) dated October 30, 1984 (Reference 13), SSES Unit 1 complies with General Design Criteria 12 by detecting and suppressing power oscillations.

7.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the U1C3 reload are described in the ENC topical report XN-NF-80-19(A), Vol. 1, and Vol. 1 Supplements 1 and 2 (Reference 14). 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 design is an 8x8 lattice with two (2) inert (water) rods and 62 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 five different enrichments. The fuel bundle burnable poison design includes six (6) 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 2.98 w/o U235 and the bundle average enrichment (including the top natural uranium blanket) is 2.89 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
15	2.48 (6 containing 4.0 w/o GD_2O_3)
21	2.94
20	3.72

The neutronic design parameters and pin enrichment distribution are described in section 4.0 of the U1C3 Reload Analysis Report (Reference 4). A brief summary of the fuel assembly reactivity (K∞) for the 2.98 w/o U235 enriched zone at various beginning-of-life (BOL) conditions is provided in the following table:

Condition	Fuel Temperature (^{o}F)	Moderator Temperature (^{o}F)	Void Fraction (%)	Uncontrolled K∞	Controlled K∞
Cold	68	68	0	1.1116	0.9573
Intermediate	180	180	0	1.1050	0.9450
Hot Standby	548.8	548.8	0	1.0795	0.8842
Hot Operating	1007	548.8	0	1.0754	0.8809
Hot Operating	1007	548.8	40	1.0509	0.8234
Hot Operating	1007	548.8	70	1.0283	0.7637

7.2 Core Reactivity

The beginning-of-cycle 3 (BOC 3) cold core Keff value with all-rods-out was calculated to be 1.11157. Based on the nominal Cycle 2 length of 5,005 MWD/MTU, a minimum Shutdown Margin of 1.99% ΔK/K, with the strongest worth control rod fully withdrawn at cold (68^{o}F) reactor conditions, was determined to occur at a Cycle 3 exposure of 7,500 MWD/MTU. The BOC 3 Shutdown Margin was calculated to be 2.26% ΔK/K. Therefore, the difference between the minimum Shutdown Margin in the cycle and the BOC Shutdown Margin, R, is

0.27% ΔK/K. The calculated Shutdown Margin is well in excess of the 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 4.67% ΔK/K with the reactor in a cold, xenon free state, and all control rods in 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; and thus for the Cycle 3 reload core, confirms the basis of the Technical Specification requirement.

7.3 Contrast of Cycle 3 Core with Cycle 2

The core loading strategies for Cycles 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. Twice burned GE Type III bundles will be utilized on the core periphery and to hold down otherwise "hot" four bundles cells. The remaining bundles, which are 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. The reference design loading pattern also maintains quarter core reflective symmetry in Cycle 3 as it did for Cycle 2.

An important difference in the design of Cycles 2 and 3 is that Cycle 3 will be the first 18 month reload Cycle for SSES Unit 1. The resultant increase in energy requirements require that a higher enriched reload bundle (2.89 w/o U235 for XN-2 versus 2.72 for XN-1) be utilized. Briefly reviewing the previous fuel bundle designs, the twice burned GE Type III fuel initially contained axially varying gadolinia at 2, 4, and 5 w/o Gd_2O_3 in the enriched zones of designated rods, while the XN-1 fuel initially contained 2 w/o Gd_2O_3 distributed uniformly over the enriched length of the designated rods. The XN-2 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 longer 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, like the XN-1 fuel design, it is not necessary to include axial varying gadolinia in the XN-2 fuel. The XN-2 and XN-1 fuel utilize similar enrichment distributions to yield internal power peaking which results in a balanced and acceptable design relative to MCPR and MAPLHGR Limits. In addition, both fuel designs contain six (6) inch natural uranium sections 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 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 found that as long as the $K\infty$ of the fuel lattice (i.e. the maximum enriched zone of the bundle) is ≤ 1.30, under dry or flooded conditions this will insure that the new fuel vault Keff remains below 0.95 as specified in the SSES FSAR.

The above mentioned K^∞ is calculated for a cold ($20^{\circ}C$), moderated, uncontrolled fuel assembly lattice in reactor geometry at beginning-of-life (BOL). The cold, uncontrolled, BOL K^∞ for the XN-2 fuel assembly enriched zone, as calculated by ENC and listed in section 7.1, is 1.1116. This value is well below the GE analysis criteria of 1.30, and 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 all times under optimum moderation conditions, 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 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.

With these assumptions, it was calculationally demonstrated that the spent fuel pool Keff would always remain below 0.95 as stipulated in the FSAR. Thus, 8x8 fuel assemblies can be safely stored as long as the average enrichment of the maximum enriched zone of the assembly is $\leq$ 3.25 w/o U235. The average enrichment of the XN-2 fuel assembly enriched zone is 2.98 w/o U235 as described in section 7.1. This enrichment is significantly lower than the 3.25 w/o criteria, and 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 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 Cycle 2. 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 14). 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 U1C3 fuel, eight categories of core-wide potential transients are considered as described in ENC's Plant Transient Methodology Report XN-NF-79-71(P) (Reference 15). ENC has provided analysis results for the following three core-wide transients to determine the thermal margin for U1C3:

1) Generator Load Rejection without Bypass (LRWOB)
2) Feedwater Controller Failure (FWCF)
3) Loss of Feedwater Heating (LOFWH).

As shown in XN-NF-79-71(P) (Reference 15), 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 14). The results of the core-wide and local transient analyses are provided in the U1C3 Reload Analysis Report XN-NF-85-132 (Reference 4) and in the U1C3 Plant Transient Analysis Report XN-NF-35-130 (Reference 5). These documents describe the correspondence between the generic documents listed above and the U1C3 specific cases. The Rod Withdrawal Error was determined to be the limiting event for U1C3 resulting in a ΔCPR of 0.19, and when combined with the 1.06 Safety Limit, requires a MCPR operating limit of 1.25.

Additional analyses were performed to determine the MCPR operating limit with a 108% rod block monitor (RBM) setpoint, the turbine bypass system inoperable, and the End-of-Cycle recirculation pump trip logic (EOC-RPT) inoperable. The resulting ΔCPRs are 0.23 for a 108% RBM setpoint, 0.20 for the bypass system inoperable, and 0.27 for the EOC-RPT inoperable. Therefore, operation with a 108% RBM setpoint requires a MCPR operating limit of 1.29; operation with the bypass system inoperable requires a MCPR operating limit of 1.26; and operation with the RPT inoperable requires a MCPR operating limit of 1.33.

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. The LRWOB event was found to be the most limiting core wide event, but resulted in a lower ΔCPR than the Rod Withdrawal Error event. Therefore, a statistical convolution of input parameters was not required and all core wide transients were analyzed deterministically (using bounding values as input parameters). The Loss of Feedwater Heater (LOFWH) event was analyzed deterministically with ENC's PTSBWR code (XN-NF-79-71 Rev. 2) which uses a point-kinetics neutronics model since rapid pressurization and void collapse do not occur for this event. Both codes utilize a multi-node steam line model to accommodate pressure waves in the steam line.

Technical Specification scram times were used in this bounding analysis. Therefore, the calculated operating limit MCPR is conservative for scram speeds less than the Technical Specification scram speeds. Therefore, no scram speed adjustment to the MCPR operating limit is required for Cycle 3 operation of SSES Unit 1.

9.2 Local Transients

As shown in XN-NF-85-132 (Reference 4), 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 0.23, and for a 106% full flow RBM setpoint the ΔCPR is 0.19.

9.3 Reduced Flow Operation

ENC has provided MCPR operating limits for manual flow control reduced flow operation for Cycle 2 in XN-NF-84-118 Supplement 1 (Reference 16). These values are based on ENC's analysis of the recirculation pump flow increase event from reduced flow operation for SSES Unit 1. These analyses are still applicable for Cycle 3 operation. The operating limit consists of a plot of MCPR versus core flow. Therefore, for all power/flow conditions, the U1C3 operating MCPR limit is the maximum of the 100% flow operating limit based on the RWE analysis and the reduced flow operating limit based on the recirculation flow increase event. Since the automatic load following capability has been removed from SSES Unit 1, analyses for the automatic flow control mode of operation were not 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 U1C3 analysis assumes six safety relief valves are out of service. The maximum pressure observed in the analysis (at the vessel bottom) is 1275 psig or 102% of reactor vessel design pressure which is well within the 110% design criterion.

The calculated steam dome pressure corresponding to the 1275 psig peak vessel pressure is 1254 psig, for a vessel differential pressure of 21 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). Since the calculated vessel differential pressure is 21 psi, the choice of 1325 psig assures compliance with the ASME criterion of 1375 psig peak vessel pressure while also maintaining consistency with the U1C3 pressure safety limit.

10.0 POSTULATED ACCIDENTS

In support of U1C3 operation, ENC has analyzed the Loss-of-Coolant Accident (LOCA) to determine MAPLHGR limits for XN-2 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-85-132 (Reference 4). ENC's methodology for the RDA analysis is described in XN-NF-80-19(A) Vol. 1 (Reference 14) 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 which defined 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 102% power and 100% flow for SSES Unit 1 is provided in XN-NF-84-119 (Reference 21). XN-NF-85-132 (Reference 4) contains LOCA calculations for the XN-2 fuel which show that the MAPLHGR limits established for the XN-1 fuel result in acceptable Peak Cladding Temperatures (PCT) for the XN-2 fuel. Therefore, operation within the MAPLHGR limits of Section 7.2.1 of XN-NF-85-132 (Reference 4) for XN-2 fuel will ensure that the peak cladding temperature (PCT) remains below 2200°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 MAPLHGR limits at bundle exposures greater than 25,000 MWD/MTU differ from the Cycle 2 limits to remain within the power history assumed in the fuel mechanical design analysis (References 7 and 8). The Cycle 2 limits were based on a non-approved mechanical design analysis (Reference 22). This did not affect Cycle 2 operation because the XN-1 fuel bundles were not expected to reach 25,000 MWD/MTU exposure, and although not required by the Technical Specifications, POWERPLEX explicitly monitors the local power history for compliance with the fuel mechanical design analysis as discussed in Section 5. However, for Cycle 3 operation the XN-1 and XN-2 MAPLHGR limits are consistent with the fuel mechanical design analysis and conservative relative to the values derived in the LOCA analysis. As shown in XN-NF-85-14 (Reference 23), a LOCA initiated from 102% power and 87% core flow (operation in the expanded power/flow region) results in a slightly lower PCT using ENC EXEM methodology. Since the 102% power/100% flow analysis results in a 2147°F PCT for XN-1 fuel and 2162°F PCT for XN-2 fuel, operation within the expanded power/flow region will not result in exceeding the 2200°F licensing limit. The LOCA analyses of XN-NF-84-119 (Reference 21) and XN-NF-85-132 (Reference 4) were performed for an entire core of XN-2 fuel and therefore provide MAPLHGR's for ENC fuel only.

As discussed in Sections 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 24 for the expanded power/flow region) and MAPLHGR limits will remain applicable during U1C3 and future cycles with GE/ENC mixed cores.

10.2 Rod Drop Accident

ENC's methodology for analyzing the Rod Drop Accident (RDA) is described in XN-NF-80-19(A) Vol. 1 (Reference 14) 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 U1C3, section 6.0 of XN-NF-85-132 (Reference 4) shows a value of 83 cal/gm for the maximum fuel rod enthalpy during the worst case postulated RDA. This value is considerably lower than the maximum fuel rod enthalpy calculated for Cycle 2 and 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 25).

REFERENCES

1) PLA-2393, "Susquehanna Steam Electric Station Proposed Amendment 59 to License No. NPF-14," Pennsylvania Power & Light Co., January 15, 1985.

2) PLA-2412, "Susquehanna Steam Electric Station Supplement to Proposed Amendment 59 to License Number NPF-14," Pennsylvania Power & Light Co., February 21, 1985.

3) Letter from W. Butler, NRC, to N. W. Curtis, PP&L, "Amendment No. 45 to Facility Operating License No. NPF-14 - Susquehanna Steam Electric Station, Unit 1," May 22, 1985.

4) XN-NF-85-132, Rev. 1, "Susquehanna Unit 1 Cycle 3 Reload Analysis," Exxon Nuclear Co., December 1985.

5) XN-NF-85-130, Rev. 1, "Susquehanna Unit 1 Cycle 3 Plant Transient Analysis," Exxon Nuclear Co., December 1985.

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) XN-NF-81-21(A), Rev. 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Co., September 1982.

8) XN-NF-81-21(P), Rev. 1, Supplement 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR 8x8 Reload Fuel," Exxon Nuclear Co., March 1985.

9) XN-NF-81-58(A), Supplements 1&2, Rev. 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Co., March 1984.

10) PLA-1263, "Response To License Condition #14," Pennsylvania Power & Light Co., August 27, 1982.

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