PL-NF-90-005

Susquehanna SES Unit 2
Cycle 5

RELOAD SUMMARY REPORT

Nuclear Fuels
Engineering

September 1990

PP&L

Pennsylvania Power & Light Company

SUSQUEHANNA SES Unit 2 Cycle 5
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NOTICE

This technical report was derived from information developed during PP&L's nuclear design and licensing analysis activities and from safety and licensing information provided to PP&L by Advanced Nuclear Fuels Corporation. This report 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 5 reload license amendment. 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

During Cycle 5 operation, Susquehanna Steam Electric Station (SSES) Unit 2 will contain the fourth reload of Advanced Nuclear Fuels Corporation 9x9 fuel in SSES Unit 2 and the second fuel and core nuclear design developed by PP&L for Unit 2. This report provides a general discussion and summary of the results of the reload analyses performed by PP&L and Advanced Nuclear Fuels Corporation (ANF) in support of SSES Unit 2 Cycle 5 (U2C5) operation. PP&L developed the fuel and core nuclear design and performed related analyses (e.g., Shutdown Margin, Hot Excess Reactivity, and cycle length determination). PP&L also performed most of the licensing analyses using methods described, benchmarked, and demonstrated in References 1, 2, and 3. The licensing analyses that PP&L performed are: Shutdown Margin; Standby Liquid Control System capability; Control Rod Drop Accident; Loss of Feedwater Heating; Rod Withdrawal Error; Fuel Loading Error (both Rotated and Mislocated); Generator Load Rejection without Bypass; Feedwater Controller Failure; Recirculation Flow Controller Failure; and, ASME Overpressure compliance. ANF provided results for the U2C5 stability, LOCA, MCPR Safety Limit type analyses, Fuel Storage Criticality, Single Loop Operation and Fuel and Equipment Handling Accidents. The PP&L analyses, evaluations, and results presented in this report are similar to those submitted in Reference 3. The ANF analyses, evaluations, and results presented in this report and the reports referenced herein are similar to those submitted in support of both SSES Unit 2 Cycle 4 operation (Reference 4) which were approved by the NRC (Reference 5) and SSES Unit 1 Cycle 6 operation (Reference 7).

Also included are a description of the U2C5 reload fuel and core design, a description and discussion of control blade replacements for U2C5, and a brief discussion of the license amendment (i.e., proposed Technical Specification changes).

The issue of core stability has been addressed for U2C5 through several calculations, previous startup tests (Section 7.4 of this report), and implementation of the interim operating guidelines presented in NRC

Bulletin 88-07 Supplement 1 via Technical Specifications. This approach is consistent with the current Unit 2 Cycle 4 method for addressing core stability. PP&L will evaluate long term solutions developed by the BWR Owner's Group Stability Committee when they are complete.

This U2C5 Reload Summary Report along with the proposed changes to the SSES Technical Specifications serve as the basic framework for the reload licensing submittal. Where appropriate, reference is made to applicable supporting documents containing more detailed information and/or specifics of the applicable analysis. The analyses performed by ANF, as listed above, were generated in compliance with ANF topical report XN-NF-80-19(P)(A), Vol. 4 Rev. 1, "Application of ENC Methodology to BWR Reloads" (Reference 6). Reference 6 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 specific reload documents and the applicable generic reload documents (methodology previously approved or currently under review) which are being used in support of the U2C5 reload core submittal.

2.0 GENERAL DESCRIPTION OF RELOAD SUBMITTAL SCOPE

During the fourth refueling and inspection outage at SSES Unit 2, PP&L will replace 232 irradiated ANF 9x9 fuel assemblies (approximately 30 percent of the previous Cycle 4 core) with 232 fresh ANF-4 9x9 fuel assemblies. The ANF-4 9x9 fuel has similar operating characteristics (thermal-hydraulic and nuclear) to the ANF-3 9x9 design which was previously approved (Reference 5). The Cycle 5 reload core required the performance of a wide range of analyses to support U2C5 core operation. These included analyses for anticipated operational occurrences and postulated accidents. In addition, the generic Pump Seizure Accident analysis submitted for SSES Unit 1 Cycle 6 (Reference 7) is being used to support Single Loop Operation (SLO) for Unit 2 Cycle 5. Analyses for normal operation of the reactor consisted of fuel evaluations in the areas of mechanical, thermal-hydraulic, and nuclear design.

Based on PP&L's design and analyses and ANF's analyses of the Cycle 5 reload core, a number of proposed changes to the SSES Unit 2 Technical Specifications have resulted. Proposed changes also exist to incorporate PP&L's Reload Licensing Analysis methodology (Reference 3). The rationale used to arrive at these proposed changes is contained in this document.

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

Proposed Changes to Technical Specifications

2.1 - Safety Limits
3/4.2.3 - Minimum Critical Power Ratio
3/4.4.1 - Recirculation System

Proposed Changes to Technical Specification Bases

2.1 - Safety Limits
3/4.1.3 - Control Rods
3/4.1.4 - Control Rod Program Controls
3/4.2.3 - Minimum Critical Power Ratio
3/4.4.1 - Recirculation System

Proposed Changes to Design Features

5.3 - Reactor Core

3.0 SSES UNIT 2 CYCLE 4 CORE OPERATING HISTORY

To date, the Cycle 4 core 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 core design analysis was used, together with projected plant operation, as a basis for the Cycle 5 core design and as input to the

reload licensing analyses. The Cycle 4 core is expected to operate within the assumptions of the Cycle 5 reload licensing analyses; therefore, the remainder of Cycle 4 core operation will not affect the licensing basis of the Cycle 5 reload core. If Cycle 4 does not operate within the assumptions of the Cycle 5 reload licensing analyses, the effects on the reload licensing analyses will be evaluated.

4.0 RELOAD CORE DESCRIPTION

The U2C5 core designed by PP&L will consist of 764 fuel assemblies, including 232 fresh ANF 9x9 assemblies (ANF-4), 204 once burned ANF 9x9 assemblies (ANF-3), 236 twice burned ANF 9x9 assemblies (XN-2), and 92 XN-1 9x9 assemblies. Of the 92 XN-1 9x9 assemblies, 85 are thrice burned, 6 are twice burned, and one is a repaired twice burned assembly. The repaired assembly was described in Reference 4 for U2C4 operation and U2C4 operation was approved by Reference 5. The six twice burned assemblies missed one cycle of irradiation because they were symmetric to failed fuel assemblies. The repaired fuel assembly failed during U2C2. The failed assembly was repaired during U2C3, and the repaired assembly and its three symmetric assemblies were returned to use in U2C4. A different fuel assembly failed during U2C3, and after inspection PP&L decided not to reuse it; however its three symmetric assemblies are being returned for use during U2C5. The ANF-4 reload fuel consists of 232 bundles which contain nine burnable poison rods with 5.0 wt% Gd_2O_3 (9Gd5) at a bundle average enrichment of 3.43 wt% U-235. A breakdown by bundle type/bundle average enrichment is provided in the following table:

Number of Bundles	Bundle Type
232	ANF 9x9/3.43 wt% U235 fresh ANF-4 (9Gd5)
100	ANF 9x9/3.17 wt% U235 once burned ANF-3 (9Gd4)
104	ANF 9x9/3.33 wt% U235 once burned ANF-3 (9Gd5)
140	ANF 9x9/3.33 wt% U235 twice burned XN-2 (9Gd4)
96	ANF 9x9/3.33 wt% U235 twice burned XN-2 (10Gd5)
85	ANF 9x9/3.31 wt% U235 thrice burned XN-1 (7Gd4)

6	ANF 9x9/3.31 wt% U235 twice burned XN-1 (7Gd4)
1	ANF 9x9/3.31 wt% U235 repaired twice burned XN-1 (7Gd4)

The anticipated Cycle 5 core loading configuration along with additional core design details is presented in Figure 1. The core is a conventional scatter loading with the lowest reactivity bundles placed in the peripheral region of the core. A minor asymmetry exists on the periphery of the core where three of the twice burned XN-1 assemblies are loaded quarter core symmetrically with a thrice burned XN-1 assembly. This is due to PP&L's decision not to reuse a fuel assembly that failed during U2C3. PP&L analyzed this asymmetry and determined that no significant effect would result on the safety analyses, core operation, or core monitoring which are based on quarter core symmetric calculations. In addition, three other twice burned XN-1 assemblies are loaded symmetrically with the repaired twice burned XN-1 assembly. The loading pattern was designed to obtain the required energy while meeting the constraints on shutdown margin, hot excess reactivity, and power peaking.

5.0 CONTROL BLADES

In response to IE Bulletin 79-26, Rev. 1, PP&L committed to replacing control blades prior to exceeding a limit of 34 percent B^{10} depletion averaged over the upper one-fourth of the control blade (Reference 8). To ensure that this limit is not exceeded during Susquehanna SES Unit 2 Cycle 5 operation as well as for other operational objectives, PP&L plans to replace up to 50 of the original equipment control blades before U2C5 operation. The original equipment control blades will be replaced with GE Duralife 160C control blades. The Duralife 160C control blade is designed to eliminate the B_4C tube cracking problem and increase the control blade assembly life. The main differences between the Duralife 160C control blades and the original equipment control blades are:

a) the Duralife 160C control blades utilize three solid hafnium rods at each edge of the cruciform to replace the three B_4C rods that are most susceptible to cracking and to increase control blade life;

b) the Duralife 160C control blades utilize improved B_4C tube material (i.e. high purity stainless steel vs. commercial purity stainless steel) to eliminate cracking in the remaining B_4C rods during the lifetime of the control blade;

c) the Duralife 160C control blades utilize GE's crevice-free structure design, which includes additional B_4C tubes in place of the stiffeners, an increased sheath thickness, a full length weld to attach the handle and velocity limiter, and additional coolant holes at the top and bottom of the sheath;

d) the Duralife 160C control blades utilize low cobalt-bearing pin and roller materials in place of stellite which was previously utilized;

e) the Duralife 160C control blade handles are longer by approximately 3.1 inches in order to facilitate fuel moves within the reactor vessel during refueling outages at Susquehanna SES; and

f) the Duralife 160C control blades are approximately 16 pounds heavier as a result of the design changes described above.

The Duralife 160C control blade has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Duralife 160C control blade stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control blade insertion capability has been evaluated and found to be acceptable during all modes of plant operation within the limits of plant

analyses. The Duralife 160C control blade coupling mechanism is equivalent to the original equipment coupling mechanism, and is therefore fully compatible with the existing control rod drives in the plant. In addition, the materials used in the Duralife 160C are compatible with the reactor environment. The impact of the increased weight of the control blades on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been evaluated and found to be negligible.

With the exception of the crevice-free structure and the extended handle, the Duralife 160C control blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (Reference 9). The mechanical aspects of the crevice-free structure were approved by the NRC for all control blade designs in Reference 10. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same methodology as was used for the Hybrid I control blades in Reference 9. These calculations were performed for the original equipment control blades and the Duralife 160C control blades described above assuming an infinite array of ANF 9x9 fuel. The Duralife 160C control blade has a slightly higher worth than the original equipment design, but the increase in worth is within the criterion for nuclear interchangeability. The increase in blade worth has been taken into account in the appropriate U2C5 analyses. However, as stated in Reference 9, the current practice in the lattice physics methods is to model the original equipment all B_4C control blade as non-depleted. The effects of control blade depletion on core neutronics during a cycle are small and are inherently taken into account by the generation of a target k-effective for each cycle. As discussed above, the neutronics calculations of the crevice-free structure show that the non-depleted Duralife 160C control blade has direct nuclear interchangeability with the non-depleted original equipment all B_4C design. The Duralife 160C also has the same end-of-life reactivity worth reduction limit as the all B_4C design. Therefore, the Duralife 160C can be used without changing the current lattice physics models as previously approved for the Hybrid I control blades (Reference 9).

The extended handle and the crevice-free structure features of the Duralife 160C control blades result in a one pound increase in the control blade weight over that of the Hybrid I blades, and a sixteen pound increase over the Susquehanna SES original equipment control blades. In Reference 9, the NRC approved the Hybrid I control blade which weighs less (by more than one pound) than the D lattice control blade. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control blade weighs less than the D lattice control blade, and the heavier D lattice control blade speed is used in the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control blade scram times are not significantly different than the original equipment control blade scram times. The current Susquehanna SES measured scram times also have considerable margin to the Technical Specification limits. Since the increase in weight of the Duralife 160C control blades does not significantly increase the measured scram speeds and the safety analyses which involve reactor scrams utilize either the Technical Specification limit scram times or a range of scram times up to and including the Technical Specification scram times, the operating limits are applicable to U2C5 with Duralife 160C control blades.

Since the Duralife 160C control blades contain solid hafnium rods in locations where the B_4C tubes have failed, and the remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steel vs. commercial purity stainless steel), boron loss due to cracking is not expected. Therefore, the requirements of IE Bulletin 79-26, Revision 1 do not apply to the Duralife 160C control blades. However, PP&L plans to continue tracking the depletion of each control blade and discharge any control blade prior to a ten percent loss in reactivity worth.

6.0 FUEL MECHANICAL DESIGN

The mechanical design and supporting analyses of the U2C5 ANF-4 fuel are the same as those for the SSES Unit 2 Cycle 4 ANF-3 fuel and are described in XN-NF-85-67(P)(A), Revision 1 (Reference 11), XN-NF-84-97 (Reference 12), PLA-2728 (Reference 13), XN-NF-82-06(P)(A), Supplement 1, Revision 2 (Reference 14). Each ANF-4 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 for the ANF 9x9 fuel design. The RODEX2, RODEX2A, RAMPEX and COLAPX codes were used in the generic mechanical design analyses. All parameters meet their respective design limits as described in Reference 11. The generic analyses for the 9x9 design (Reference 11) are applicable to the XN-1, XN-2, ANF-3, and ANF-4 fuel designs and support a maximum 9x9 assembly discharge exposure of 40,000 MWD/MTU. Based on calculations, U2C5 operation is projected to result in a peak 9x9 assembly exposure less than 40,000 MWD/MTU.

For the ANF 9x9 fuel, the design is such that adequate margins to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) are assured for all anticipated operational occurrences throughout the life of the fuel as demonstrated by the fuel design analyses (Reference 11), provided that the fuel rod power history remains within the power histories assumed in the analyses. The steady state design power profile for the ANF 9x9 fuel is shown in Figure 3.3 of Reference 11. This power profile 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 Thermal Power divided by Maximum Fraction of Limiting Power Density (FRTP/MFLPD) was incorporated. This ensures

that ANF 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 9x9 fuel is based on ANF's Protection Against Fuel Failure (PAFF) line as shown in Figure 3.4 of Reference 11 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 AOOs.

The overall structural response of the ANF 9x9 assembly design during Seismic-LOCA events is essentially the same as the response of the GE 8x8R assembly design that comprised the initial Susquehanna SES Unit 1 core. The similar physical properties and bundle natural frequencies result in nearly identical structural responses as discussed in previous submittals (Reference 4). In addition, Reference 11 presents the ANF 9x9 fuel assembly component Seismic-LOCA analysis which showed large design margins to the fuel design limits. Additional justification (Reference 13) was also provided to the NRC by PP&L during the Unit 2 Cycle 2 reload licensing process.

7.0 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 assemblies in the core, (2) MCPR Safety Limit type analyses, (3) bypass flow characteristics, and (4) thermal-hydraulic stability. The analyses performed to determine each of these parameters are discussed in this section.

7.1 Hydraulic Compatibility

Component hydraulic resistances for all Unit 2 Cycle 5 fuel are the same for all reload fuel and have been determined in single phase flow tests of full scale assemblies. Thermal hydraulic

compatibility is assured because the Unit 2 Cycle 5 core loading is entirely ANF 9x9 fuel.

7.2 MCPR Safety Limit type analysis

The PP&L Statistical Combination of Uncertainties (SCU) methods are described in Reference 3. When using the SCU methodology, the transient ΔCPR and traditional MCPR safety limit analyses are combined into a single unified analysis. As a result, the high pressure, high flow safety limit is not represented as a single MCPR value, but rather as a condition such that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. As described in Appendix B of Reference 3, a set of "MCPR Safety Limit type" analyses are performed for several values of MCPR. The MCPR Safety Limit type analyses were performed by ANF using the same methods and assumptions as the traditional MCPR Safety Limit analysis.

As shown in Table 1, a MCPR value of 1.06 in two loop operation assures that less than 0.1% of the fuel rods are expected to experience boiling transition. The methodology and generic uncertainties used in the MCPR Safety Limit type calculations are provided in XN-NF-80-19(P)(A), Volume 4 Revision 1 (Reference 6). The uncertainties used for the SSES U2C5 MCPR Safety Limit Type calculations are the same as for U2C4 and are presented in Reference 18. The results are presented in Table 1.

During U2C5, as in the previous cycle, the ANF 9x9 fuel will be monitored using the XN-3 critical power correlation. ANF has determined that this correlation provides sufficient conservatism to preclude the need for any penalty due to channel bow during U2C5. Susquehanna SES is a C-lattice plant and uses channels for only one fuel bundle lifetime. The conservatism has been evaluated by ANF to be greater than the maximum expected ΔCPR (0.02) due to channel bow in C-lattice plants using channels for only one fuel bundle

lifetime. Therefore, the monitoring of the MCPR limit is conservative with respect to channel bow and addresses the concerns of NRC Bulletin No. 90-02 (Reference 16). The details of the evaluation performed by ANF have been reported generically to the NRC (Reference 17).

7.3 Core Bypass Flow

Core bypass flow is calculated using the methodology described in PL-NF-87-001-A (Reference 1). The core bypass flow fraction (including water rod flow) for U2C5 is 8.7% of total core flow which is the same as the Cycle 4 bypass flow value of 8.7%. The bypass flow fraction is used in the MCPR Safety Limit type calculations and as input to the cycle specific transient analyses.

7.4 Core Stability

COTRAN core stability calculations were performed for Unit 2 Cycle 5 to determine the decay ratios at predetermined power/flow conditions. The resulting decay ratios were used to define operating regions which comply with the interim requirements of NRC Bulletin No. 88-07, Supplement 1 "Power Oscillations in Boiling Water Reactors," (Reference 19). As in the previous cycle, Regions B and C of the NRC Bulletin have been combined into a single region (i.e., Region II), and Region A of the NRC Bulletin corresponds to Region I.

Region I has been defined such that the decay ratio for all allowable power/flow conditions outside of the region is less than 0.90. To mitigate or prevent the consequences of instability, entry into this region requires a manual reactor scram. Region I for Unit 2 Cycle 5 has been calculated to be slightly different than Region I for the previous cycle.

Region II has been defined such that the decay ratio for all allowable power/flow conditions outside of the region (excluding Region I) is less than 0.75. For Unit 2 Cycle 5, Region II must be immediately exited if it is inadvertently entered. Similar to Region I, Region II is slightly different than in the previous cycle.

In addition to the region definitions, PP&L has performed stability tests in SSES Unit 2 during initial startup of Cycles 2, 3 and 4 to demonstrate stable reactor operation with ANF 9x9 fuel. The test results for U2C2 (Reference 20) show very low decay ratios with a core containing 324 ANF 9x9 fuel assemblies. Analysis of data taken during U2C2 Two Loop Operation at 60% power and 47% flow resulted in a "measured" decay ratio of 0.33 and a COTRAN calculated decay ratio of 0.33. In Single Loop Operation at 55% power and 44% flow the "measured" decay ratio was 0.30 and the COTRAN calculated value was 0.29. In addition, the use of the ANF "ANNA" software to analyze APRM signals from the U2C3 startup produced a "measured" decay ratio of approximately 0.37 at 60% power and 46% flow. The U2C3 core contained 556 ANF 9x9 assemblies. The U2C4 core contains 764 (full core) ANF 9x9 assemblies. Two loop stability tests similar to those described above were performed at BOC 4 and the test data has been sent to the NRC (Reference 21). Stability tests are not planned for U2C5.

PP&L believes that the use of Technical Specifications that comply with NRC Bulletin 88-07 Supplement 1, and the tests and analyses described above, will provide assurance that SSES Unit 2 Cycle 5 will comply with General Design Criteria 12, Suppression of Reactor Power Oscillations. This approach is consistent with the SSES Unit 2 Cycle 4 method for addressing core stability (References 4 and 5).

8.0 NUCLEAR DESIGN

The neutronic methods for the design and analysis of the U2C5 reload are described in PP&L topical reports PL-NF-87-001-A, PL-NF-89-005, and PL-NF-90-001 (References 1, 2, and 3), ANF topical reports XN-NF-80-19(A), Vol. 1, and Vol. 1 Supplements 1 and 2 (Reference 22), and ANF letter RAC:058:88 (Reference 23). These reports have been reviewed and approved by the Nuclear Regulatory Commission for application to the Susquehanna SES reloads, except for PL-NF-89-005 and PL-NF-90-001 which are being reviewed by the NRC.

8.1 Fuel Bundle Nuclear Design

The ANF-4 fuel bundle design is 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 eight different enrichments. The ANF-4 reload batch consists of 232 bundles which contain nine burnable poison rods with 5.0 wt% Gd_2O_3 (9Gd5) blended with UO_2 enriched to 3.40 wt% U-235. The Gd_2O_3-UO_2 rods are utilized to reduce the initial reactivity of the bundle.

The average enrichment of the enriched zone is 3.54 wt% U235 for the lattice containing 9Gd5. The corresponding bundle average enrichment (including the top natural uranium blanket) is 3.43 wt% U235. The number of fuel rods at each enrichment is given below:

3.54 wt% U235
Lattice with 9Gd5

Rod Enrichment (wt% U235)	# of Rods
2.00	1
2.20	3
2.40	2
2.70	15

3.50	21
3.94	13
4.70	15
3.40	9 (5 wt% GD_2O_3)

The neutronic design parameters and pin enrichment distribution are presented in Table 2 and Figure 2, respectively.

8.2 Core Reactivity

Shutdown Margin for U2C5 was analyzed using PP&L's core physics methods (References 1 and 3) and a low Cycle 4 exposure of 9,601 MWD/MTU, which results in a conservatively high cold core reactivity condition during Cycle 5. Shutdown Margin is defined as the core reactivity with all control rods fully inserted, except for the strongest worth control rod, at 68°F and xenon-free conditions. The minimum value of Shutdown Margin occurs at 10,125 MWD/MTU and is 1.093% $\Delta k/k$. The cold all-rods-in core k-effective at 10,125 MWD/MTU is 0.96038. The value of R, which is the difference between the BOC Shutdown Margin and the minimum Shutdown Margin during the cycle, is 0.036% $\Delta k/k$. 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 sufficient Shutdown Margin will be verified by test at BOC 5.

The Standby Liquid Control System, which is designed to inject a quantity of sodium pentaborate solution that produces a boron concentration of no less than 660 ppm in the reactor core within approximately 90 to 120 minutes after initiation, was calculated by PP&L to provide a margin of shutdown of at least 2.7% $\Delta k/k$ with the reactor in a cold, xenon free state, and all control rods at their critical full power positions. This calculation process is described in Reference 3. 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 5 reload core the basis of the Technical Specification requirement is met.

8.3 Contrast of Cycle 5 Core with Cycle 4

The core loading strategies for Cycles 4 and 5 are very similar in nature. Cycle 4 utilized a conventional scatter loading with the lowest reactivity bundles placed in the peripheral region of the core. Cycle 5 will also be based on this scatter loading principle. Fresh reload bundles will be scatter loaded in control cells throughout the core except on the core periphery. Thrice burned XN-1 bundles and twice burned XN-2 bundles will be utilized on the core periphery. Twice burned XN-2 bundles will also be used to constrain reactivity in interior control cells. The once burned ANF-3 and fresh ANF-4 bundles will be distributed throughout the core in a manner which yields acceptable radial peaking and provides adequate cold shutdown margin throughout the cycle.

Briefly reviewing the previous reload fuel bundle designs that will remain in the core for U2C5 (which are all ANF 9x9), the Cycle 2 XN-1 fuel initially contained 4 wt% Gd_2O_3 distributed uniformly over the enriched zones of seven designated rods. The Cycle 3 XN-2 and Cycle 4 ANF-3 fuel initially contained both 4 and 5 wt% Gd_2O_3 distributed uniformly over the enriched zones of designated rods in selected subbatches. The Cycle 5 ANF-4 fuel bundle design has a 3.43 wt% U235 bundle average enrichment and contains 9 gadolinia bearing rods at 5 wt% Gd_2O_3.

For reload cycles, the axial exposure profile of the exposed bundles provides an axial shaping effect and decreases the need for axial varying gadolinia in U2C5 . Thus, as was the case for the XN-1, XN-2, and ANF-3 fuel designs, it is not necessary to include axial varying gadolinia in the ANF-4 fuel for the purposes of hot operating power shape control. The ANF-4 fuel utilizes an enrichment distribution to yield a lattice internal power distribution which results in a balanced and acceptable design

relative to MCPR, MAPLHGR, and LHGR Limits. In addition, the XN-1, XN-2, ANF-3, and ANF-4 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.

8.4 New Fuel Storage Vault/Spent Fuel Pool Criticality

8.4.1 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. 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 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 ≤ 3.95 wt% U235 and an ANF calculated $k_\infty \leq 1.388$ will yield a new fuel vault k-effective $\leq .95$ under dry or flooded conditions (Reference 24).

The above mentioned k_∞ 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_∞ of the ANF-4 fuel assembly enriched zone, as calculated by PP&L is 1.112. This value is well below the ANF analysis criterion of 1.388. Thus for the ANF-4 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 (between dry and flooded), watertight covers are used, administrative procedures are in place to prevent this condition, and criticality monitors have been installed as an added precaution.

8.4.2 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 wt% U235 containing no burnable poison. The absence of burnable poisons ensures 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 ≤ .95 (Reference 25). The resulting criterion is that the average enrichment of the maximum enriched zone of a 9x9 assembly be ≤ 3.95 wt% U235. The enrichment of the enriched zone of the ANF-4 9x9 fuel design is 3.54 wt% U235. This enrichment is less than the 3.95 wt% U235 requirement, and thus it is concluded that adequate margin exists to prevent spent fuel pool criticality throughout the ANF-4 fuel assembly lifetime.

9.0 <u>CORE MONITORING SYSTEM</u>

The POWERPLEX® core monitoring system will be utilized to monitor reactor parameters during Cycle 5. POWERPLEX incorporates ANF's core simulation methodology and is used for both on-line core monitoring and as an off-line predictive and backup tool. POWERPLEX input will be based on the

CPM-2/PPL methodology (Reference 3). This methodology has been submitted to the NRC by PP&L.

The POWERPLEX system has been operational at SSES and utilized to monitor reactor parameters during Unit 1 Cycles 2, 3, 4, and 5 and Unit 2 Cycles 2, 3, and 4. The POWERPLEX routines are fully consistent with ANF's nuclear analysis methodology (with the exception of CPM-2/PPL input) as described in XN-NF-80-19(A) Volume 1 and Volume 1 Supplement 2 (Reference 22) and supplemented with the VHIST13 void history correlation (Reference 23). In addition, the measured power distribution and monitoring related uncertainties are incorporated into the MCPR Safety Limit type calculations as described in ANF's Nuclear Critical Power Methodology Report XN-NF-524(A) (Reference 18). The use of CPM-2/PPL to generate input to the POWERPLEX routines required PP&L to evaluate the monitoring related uncertainties based on the use of CPM-2/PPL. These uncertainties were determined to be less than or equal to the current monitoring related uncertainties in order to maintain the validity of the assumptions in the MCPR Safety Limit type calculations.

10.0 ANTICIPATED OPERATIONAL OCCURRENCES

The MCPR operating limits for U2C5 were generated with the PP&L reactor analysis methods described in PL-NF-90-001 (Reference 3). The U2C5 MCPR operating limits are presented as MCPR versus Percent of Rated Core Flow and MCPR versus Percent Core Thermal Power. These limits cover the allowed operating range of power and flow. As specified in PL-NF-90-001, six major events were analyzed. These events can be divided into two categories: core wide transients and local transients. The core wide transient events analyzed were:

1) Generator Load Rejection Without Bypass (GLRWOB),

2) Feedwater Controller Failure (FWCF),

3) Recirculation Flow Controller Failure - Increasing Flow (RFCF), and

4) Loss of Feedwater Heating (LOFWH)

As discussed in PL-NF-90-001, the other core wide transients are non-limiting (i.e., would produce lower calculated ΔCPRs than one of the four events analyzed). The local transient events analyzed were:

1) Rod Withdrawal Error (RWE), and

2) Fuel Loading Error (FLE).

The fuel loading error evaluation includes analysis of both rotated and mislocated fuel assemblies.

Sufficient analyses were performed to define the MCPR operating limits as a function of core power and core flow. Analyses were also performed to determine MCPR operating limits for three plant equipment availability conditions: 1) Turbine Bypass and EOC-RPT operable, 2) Turbine Bypass inoperable, and 3) EOC-RPT inoperable.

10.1 Core-Wide Transients

The PP&L RETRAN model and methods described in PL-NF-89-005 and PL-NF-90-001 (References 2 and 3) were used to analyze the GLRWOB, FWCF, and RFCF events. The ΔCPRs were evaluated using the XN-3 Critical Power Correlation (Reference 26) and the methodology described in PL-NF-90-001 (Reference 3). The GLRWOB and FWCF were analyzed in two different ways (as described in PL-NF-90-001):

1) Deterministic analyses using the Technical Specification scram speed (minimum allowed);

2) Statistical Combination of Uncertainty (SCU) analyses at an average scram speed of 4.2 feet/second.

Thus, the Technical Specification MCPR operating limits calculated for U2C5 will be a function of scram speed.

The LOFWH was conservatively analyzed by PP&L using the steady state core physics methods and process described in PL-NF-90-001, and the LOFWH results were found to be bounded by results of the other three core wide transients. The minimum MCPR operating limit required for the U2C5 LOFWH event is 1.17.

Results of the GLRWOB, FWCF, and RFCF events are presented in Tables 3, 4, and 5, respectively.

10.2 Local Transients

The fuel loading error (rotated and mislocated bundle) and the Rod Withdrawal Error (RWE) were analyzed using the methodology described in PL-NF-90-001 (Reference 3). The results of these analyses apply to all three plant equipment availability conditions previously described in Section 10, and the results are independent of scram speed. The RWE analysis supports the use of both the Duralife 160C control blades and a Rod Block Monitor setpoint of 108%. The MCPR operating limits that result from the analyses of these events are presented in Table 6. These events are non-limiting for U2C5.

10.3 ASME Overpressurization Analysis

In order to demonstrate compliance with the ASME Code overpressurization criterion of 110% of design pressure, the MSIV closure with failure of the MSIV position switch scram was analyzed by PP&L using the methods described in PL-NF-90-001. The U2C5 analysis assumed that six safety relief valves were out of service and the MSIV closure time was 2.0 seconds, which is conservative compared to the current Technical Specification minimum closure time of 3.0 seconds.

The reactor vessel components whose design pressure is 1250 psig showed the closest approach to the 110% ASME Code criterion (i.e., 1375 psig). The maximum calculated pressure in this category was

1325.3 psig, which corresponds to a margin of 49.7 psi to the limit.

11.0 POSTULATED ACCIDENTS

Three types of accidents were evaluated during the Unit 2 Cycle 5 analysis effort: the Loss of Coolant Accident (LOCA), the Control Rod Drop Accident (CRDA), and the Fuel and Equipment Handling Accidents. ANF has analyzed the Loss-of-Coolant Accident to determine the MAPLHGR limits for the ANF 9x9 fuel that will comprise the Unit 2 Cycle 5 core. PP&L generated and verified the appropriate LOCA analysis inputs as described in PL-NF-90-001 (Reference 3). ANF's methodology for the LOCA analysis is provided in References 27 through 29. PP&L performed the Control Rod Drop Accident analysis to demonstrate compliance with the 280 cal/gm Design Limit as described in PL-NF-90-001 (Reference 3) using ANF's methodology for the CRDA analysis as described in XN-NF-80-19(A) Vol. 1 (Reference 22). ANF performed an evaluation of the Fuel and Equipment Handling Accidents which are discussed in Section 11.3.

11.1 Loss-of-Coolant Accident

XN-NF-84-117(P) (Reference 30) describes ANF's generic jet pump BWR-4 LOCA break spectrum analysis. This analysis 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. The discharge coefficient assumed was 0.4, which is equivalent to a total break area of 2.8 ft^2. The analysis of this event for SSES 9x9 fuel is provided in XN-NF-86-65 (Reference 31). 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 generated by ANF are bounding for reactor operating conditions up to 100% rated power and 100% rated flow and assure acceptable peak cladding temperatures for all ANF 9x9 fuel during a postulated LOCA event. The LOCA analysis of XN-NF-86-65 (Reference 31) was

performed for an entire core of 9x9 fuel and therefore provides MAPLHGR limits for ANF 9x9 fuel only.

The generation of the local power distribution input to the heatup calculations and verification of parameters important to the blowdown calculation were performed for U2C5 by PP&L in accordance with the methodology described in PL-NF-90-001 (Reference 3). This verification determined that the blowdown calculation results are conservative for U2C5. ANF confirmed that the MAPLHGR limits in XN-NF-86-65 ensure that the Peak Cladding Temperature (PCT) for the U2C5 ANF-4 fuel 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 MAPLHGRs and PCTs for fuel resident in the U2C5 core are presented in Table 7.

11.2 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 22) and utilizes a generic parametric analysis which calculates the fuel enthalpy rise during postulated CRDAs over a wide range of reactor operating conditions. PP&L generated the parameters used in the CRDA analysis as described in PL-NF-90-001 (Reference 3). The U2C5 analysis was performed using bounding assumptions similar to those used in the U2C4 analysis presented in Reference 4. The U2C5 analysis also supported the use of the Duralife 160C control blades. For U2C5, the CRDA analysis resulted in a value of 209 cal/gm for the maximum fuel rod enthalpy and less than 640 fuel rods exceeding 170 cal/gm during the worst case postulated CRDA. The 209 cal/gm value is well below the design limit of 280 cal/gm and less than 640 fuel rods exceeding 170 cal/gm is bounded by the 770 rods assumed in Section 15.4.9 of the SSES FSAR (Reference 32). To ensure compliance with the CRDA analysis assumptions, control rod sequencing below 20% core thermal power must comply with GE's Banked Position Withdrawal Sequence constraints (Reference 33).

11.3 Fuel and Equipment Handling Accidents

Two accident analyses were performed to determine the offsite dose to the whole body and thyroid at the site boundary resulting from the dropping of an object onto the core. In the Fuel Handling Accident, the dropped object is an irradiated fuel assembly plus channel, grapple head and mast weighing a total of 1000 pounds which falls from a height of 32.95 feet above the core. In the Equipment Handling Accident, the dropped object is a mass weighing 1100 pounds which falls from a height of 150 feet above the core. The 32.95 feet represents the highest that an irradiated fuel assembly can be carried over the core; the 1100 pound mass is the largest object that is not specifically evaluated as a heavy load; and the 150 feet represents the maximum height that the overhead crane can carry an object over the core. For each of the two accidents analyzed, the number of failed fuel rods was determined and the subsequent radiological releases and offsite doses were calculated.

The number of failed fuel rods for the two cases is determined from the energy of the dropped assemblage and the energy required to fail a fuel rod. The energy required to fail a fuel rod is based upon a uniform 1% plastic deformation of the cladding. For conservatism, the minimum material properties for zircaloy-2 are used. For the Fuel Handling Accident analysis, all fuel rods in the dropped assembly are assumed to fail. For the fuel assemblies hit by the dropped assemblage in both analyses, the standard fuel rods and the tie rods are assumed to have the same failure threshold. The energy of the dropped assemblage falling from the vertical position to its side position is included in the calculation. One half of the energy is assumed to be absorbed by the falling fuel assembly and no energy is assumed to be absorbed by the 1100 pound object. For conservatism, no energy is assumed to be absorbed by the fuel pellets. The number of failed fuel rods for the Fuel Handling Accident event is 121 and for the Equipment Handling Accident event the number of failed fuel rods is 318.

The offsite dose calculations were performed assuming (1) the fission product inventories calculated by the ORIGEN computer code (Reference 37) increased by a factor of 1.5, (2) the accident occurs 24 hours after reactor shutdown, (3) the fission gas release fractions are obtained from Regulatory Guide 1.25, (4) the fuel pool decontamination factor is 100 for iodine and 1 for noble gases, (5) the standby gas treatment system removal efficiency is 99% for iodine, and (6) the atmospheric dispersion factor, breathing rate factor, and dose conversion factors are equal to those used in Chapter 15.7.4 of the Susquehanna SES FSAR.

For each of the two handling accidents analyzed, the results are shown in Table 8. As shown in Table 8 the Fuel and Equipment Handling Accident calculated doses are much less than 25% of the 10CFR100 limits.

12.0 SINGLE LOOP OPERATION

To support single loop operation for U2C5, ANF performed MCPR Safety Limit calculations considering single loop operation power/flow conditions and associated single loop operation uncertainties. The results show that the MCPR Operating Limit must be increased by 0.01 when in single loop operation. The 0.01 increase in the Operating Limit is a result of the increased measurement uncertainties associated with single loop operation.

ANF performed a review of the two loop operation limiting anticipated operational occurrences considering single loop operation. Previous analyses (References 34 and 35) indicated that other events which could be affected by single loop operation were non-limiting when analyzed under single loop operating conditions. Under single loop operating conditions, steady state operation can not exceed approximately 76% power and 60% core flow because of the capability of the operating recirculation pump. Thus, it was determined that when operating at low power/flow conditions, the two loop operation anticipated operational occurrences remain limiting. The two loop MCPR operating limits plus

0.01 conservatively protect the fuel from any transient in single loop operation.

It was determined that the single loop operation LOCA analysis presented in XN-NF-86-125 (Reference 36) is bounded by the two loop LOCA analysis. In addition, ANF analyzed the pump seizure accident from single loop operating conditions on a generic basis for the Susquehanna Units (Reference 7). The results of the generic analysis show that single loop operation of the Susquehanna Units with single loop MCPR operating limits protects against the effects of the pump seizure accident. That is, for operation at the single loop operating MCPR limit, the radiological consequences of a pump seizure accident from single loop operating conditions are but a small fraction of the 10CFR100 guidelines. Previous analyses (Reference 34) have shown that other accidents which could be affected by single loop operation were non-limiting when analyzed under single loop operating conditions.

Based on the vessel internal vibration analysis performed by GE, the 80% recirculation pump speed restriction, previously discussed in Reference 34, will be maintained for U2C5 single loop operation.

The results discussed previously in Section 7.4 on core stability also apply under single loop operating conditions. One of the stability tests performed during the startup of Susquehanna SES Unit 2 Cycle 2 was performed under single loop operating 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 test data, the stability calculation results for U2C5, and the U2C5 Technical Specifications which comply with NRC Bulletin 88-07, Supplement 1 support single loop operation during U2C5 .

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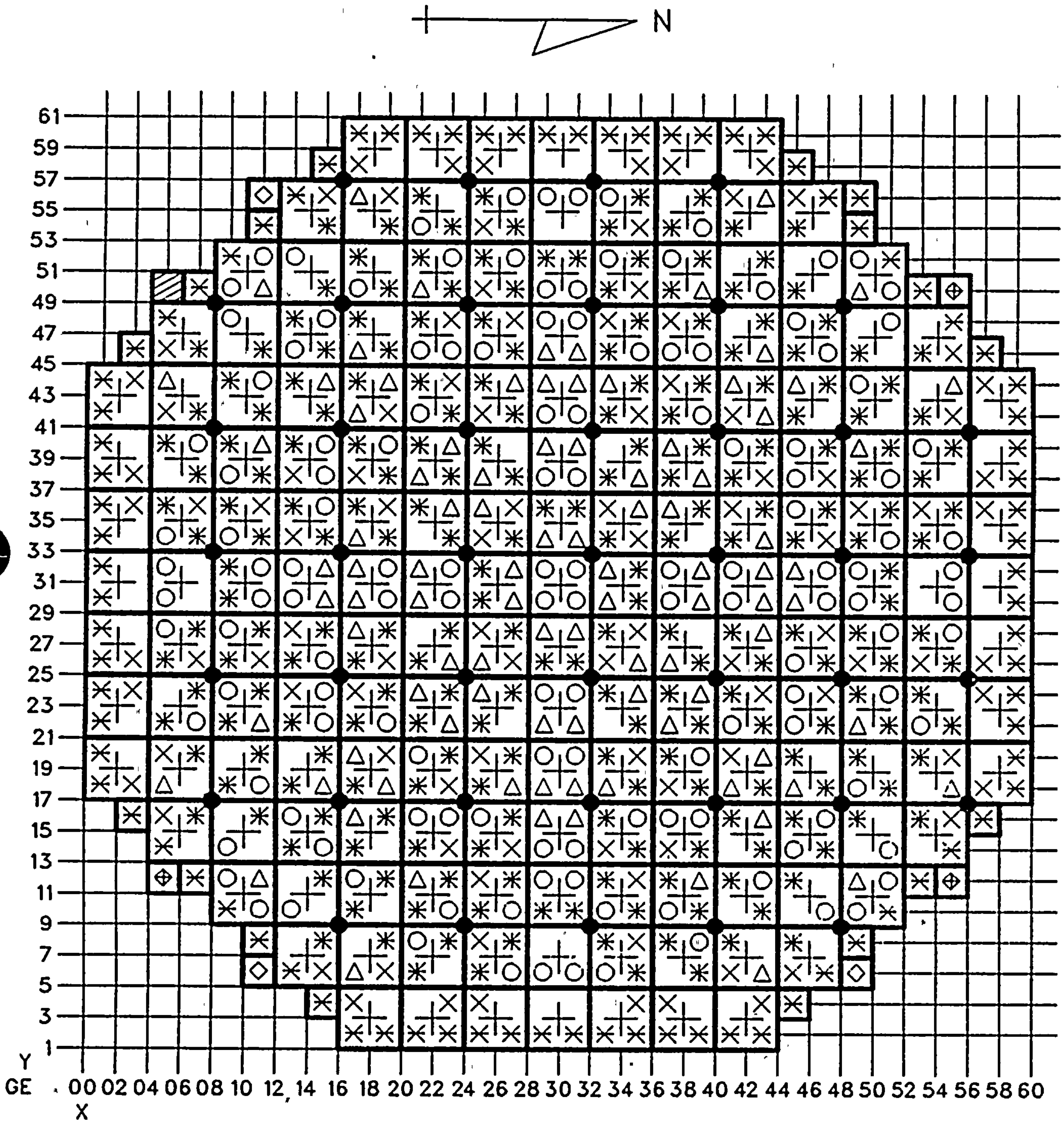
rp0192i.jhe:el

FIGURE 1

SSES UNIT 2, CYCLE 5
CORE LOADING PATTERN



ANF 9X9/XN-1 (3.31/7GD4)-85

ANF 9X9/XN-2 (3.33/9GD4)-140

ANF 9X9/XN-2 (3.33/10GD5)-96

ANF 9X9/ANF-3 (3.33/9GD5)-104

ANF 9X9/ANF-3 (3.17/9GD4)-100

ANF 9X9/XN-1 REINSERT (3.31/7GD4)-3

ANF 9X9/ANF-4 (3.43/9GD5)-232

ANF 9X9/XN-1 REINSERT (3.31/7GD4)-4

REPAIRED-1 SYMMETRICS-3

FIGURE 2

LLL	L	ML	M	M	M	ML	ML	LL
L	ML	M	MH	M*	MH	M*	M	ML
ML	M	M*	H	H	H	MH	M	ML
M	MH	H	H	H	H	H	M*	M
M	M*	H	H	W	MH	H	MH	M
M	MH	H	H	MH	W	MH	M*	M
ML	M*	MH	H	H	MH	MH	M	ML
ML	M	M	M*	MH	M*	M	M	ML
LL	ML	ML	M	M	M	ML	ML	LL

LLL RODS (1) - 2.00 w/o U-235
LL RODS (3) - 2.20 w/o U-235
L RODS (2) - 2.40 w/o U-235
ML RODS (15) - 2.70 w/o U-235
M RODS (21) - 3.50 w/o U-235
MH RODS (13) - 3.94 w/o U-235
H RODS (15) - 4.70 w/o U-235
M* RODS (9) - 3.40 w/o U-235 + 5.0 w/o Gd2O3
W RODS (2) - Inert Water Rod

U2C5 ANF-4 3.54 wt% U235 Lattice Enrichment Distribution

TABLE 1

UNIT 2 CYCLE 5
MCPR SAFETY LIMIT TYPE ANALYSES

	MCPR VALUE	PERCENT OF RODS IN BOILING TRANSITION
TWO-LOOP OPERATION	0.96	1.405%
	0.99	0.748%
	1.02	0.313%
	1.06	0.097%
	1.10	0.024%
SINGLE-LOOP OPERATION	1.07	0.079%

TABLE 2

NOMINAL SSES OPERATING CONDITIONS

Core Thermal Power	3293 MWt
Total Core Flow	100 Mlb/hr
Reactor Pressure	1020 psia
Core Inlet Subcooling	24.0 Btu/lbm
Number of Fuel Assemblies	764
Number of Control Rods	185

TABLE 3

U2C5 CALCULATED MCPR OPERATING LIMITS
GENERATOR LOAD REJECTION W/O BYPASS

Mode of Operation	Deterministic Analysis Technical Specification Scram Speed	SCU Analysis 4.2 ft/sec Scram Speed
Bypass & EOC-RPT Operable	1.47	1.32
Bypass Inoperable	1.47	1.32
EOC-RPT Inoperable	1.54	1.35

TABLE 4

U2C5 CALCULATED MCPR OPERATING LIMITS
FEEDWATER CONTROLLER FAILURE

Mode of Operation	Power (% rated)	Deterministic Analysis Technical Specification Scram Speed	SCU Analysis 4.2 ft/sec Scram Speed
Bypass & EOC-RPT Operable	100	1.31	<1.31(1)
	84	1.37	1.32
	65	1.50	1.41
	40	1.73	1.55
Bypass Inoperable	100	1.56	1.38
	84	1.64	1.43
	65	1.77	1.54
	40	1.91	1.63
EOC-RPT Inoperable	100	1.38	1.25
	84	1.41	1.31
	65	1.53	1.42
	40	1.76	1.57

(1) Not an SCU analysis, value used from deterministic analysis at Technical Specification Scram speed.

TABLE 5

U2C5 CALCULATED MCPR OPERATING LIMITS
RECIRCULATION FLOW CONTROLLER FAILURE

Core Flow (% Rated)	Calculated MCPR[1] Operating Limit
30	1.83
37	1.69
45	1.57
60	1.43
74.9	1.32

(1) Conservatively analyzed at Technical Specification scram speed. Results apply to all 3 modes of operation (i.e., Bypass and EOC-RPT operable, Bypass inoperable, and EOC-RPT inoperable).

TABLE 6

U2C5 CALCULATED MCPR OPERATING LIMITS
LOCAL TRANSIENTS

Event	Calculated MCPR Operating Limit
Rod Withdrawal Error	1.27
Mislocated Bundle	1.22
Rotated Bundle	1.28

TABLE 7

UNIT 2 CYCLE 5 LOCA HEATUP RESULTS

Limiting Break: Double-ended guillotine pipe break,
Recirculation pump discharge line,
0.4 discharge coefficient.

Assembly Average Exposure (GWd/MTU)	MAPLHGR (KW/FT)	Peak Clad Temperature (Degree F)			Peak Local MWR[1] (Percent)		
		XN-1&2[2]	ANF-3	ANF-4[3]	XN-1&2	ANF-3	ANF-4
0	10.2	2060	1998	-	3.9	2.6	-
5	10.2	2069	1937	-	3.7	1.4	-
10	10.2	2121	2079	-	3.7	3.1	-
15	10.2	2140	2126	-	4.8	4.4	-
20	10.2	2173	2161	2102	5.2	5.0	4.03
25	9.6	2016	1996	-	2.7	2.5	-
30	8.9	1839	1831	-	1.0	1.0	-
35	8.2	1752	1744	-	0.7	0.7	-
40	7.5	1676	1670	-	0.5	0.5	-

[1] Metal water reaction.

[2] Peak clad temperatures and metal water reactor (MWR) shown are bounding for ANF-9x9 XN-1 and XN-2 fuel in Susquehanna Unit 2.

[3] The ANF-4 fuel type is similar to the ANF-3 fuel type loaded in Cycle 4 except that it is slightly more edge peaked at the limiting exposure point for PCT and MWR. This exposure point was analyzed for the ANF-4 fuel type to confirm that the ANF-3 fuel peak cladding temperatures are bounding.

TABLE 8

FUEL AND EQUIPMENT HANDLING ACCIDENT RESULTS

Two-Hour Site Boundary Radiological Dose	10CFR100 Limits (Rem)	25% of 10CFR100 Limits (Rem)	Fuel Handling Accident Results (Rem)	Equipment Handling Accident Results (Rem)
Whole Body Dose	25	6	1.31	3.40
Thyroid Dose	300	75	1.81	4.74