4.3 NUCLEAR DESIGN

The nuclear design of the initial cores for Susquehanna is described in References 4.3-1, 4.3-2, and 4.3-3. This section incorporates much of the general nuclear design information in Reference 4.3-1 and presents detailed design information for reload cores.

4.3.1 Design Bases

Nuclear design bases fall into two categories: safety design bases and core performance design bases. Safety design bases are required by the General Design Criteria to ensure safe operation of the core. Core performance design bases are required to meet power production objectives.

4.3.1.1 Safety Design Bases

Safety design bases protect the nuclear fuel from damage which would result in a release of radioactivity, representing an undue risk to the health and safety of the public.

Safety design bases are listed below.

1) The core shall be capable of being rendered subcritical at any time or core condition with the highest worth control rod fully withdrawn.

2) The void coefficient shall be negative over the entire operating range.

3) Technical specification limits on Linear Heat Generation Rate (LHGR), Minimum Critical Power Ratio (MCPR), and the Average Planar Linear Heat Generation Rate (APLHGR), shall not be exceeded during steady state operation.

4) The nuclear characteristics of the design shall not exhibit any tendency toward divergent operation.

5) Reload fuel lattice enrichments shall be such that the nuclear design bases are met for the new fuel storage racks (section 9.1.1.1.1.2) and spent fuel storage (section 9.1.2.1.1.2).

4.3.1.2 Plant Performance Design Bases

1) The core design shall have adequate excess reactivity to reach the desired cycle length.

2) The core design shall be capable of operating without exceeding technical specification limits.

3) The core and fuel design and the reactivity control system shall allow continuous, stable regulation of reactivity.

4) The core and fuel design shall have adequate reactivity feedback to facilitate normal operation.

4.3.2 Description

A general description of BWR nuclear characteristics is provided in Reference 4.3-1. A summary of reactor core characteristics for Susquehanna is listed in Table 4.3-1.

4.3.2.1 Nuclear Design Description

The nuclear design of Susquehanna is both unit and cycle specific. A detailed description of the initial core nuclear design is available in Reference 4.3-1. Susquehanna Steam Electric Station Units 1 and 2 operate at power conditions in Table 4.3-1 with increased core flow. Fuel bundle and core reload designs have been developed using NRC approved methods.

4.3.2.1.1 Core Composition

The core contains 764 fuel assemblies arranged in a conventional scatter loaded pattern. Typically, the lowest reactivity fuel assemblies are placed in the peripheral region of the core. Beginning with cycles U2C21 and U1C23, ATRIUM-11 is the primary fuel type loaded into the core. During the fuel type transition cycles including U1C23, U1C24, U2C21, and U2C22, the cores will contain both ATRIUM-10 and ATRIUM-11 fuel types. In addition, a limited number of Lead Use Assemblies (LUAs) may be loaded to evaluate new fuel designs.

Detailed core compositions and associated core loading patterns are presented in References 4.3-17 and 4.3-18 for Units 1 and 2, respectively. The typical core loading patterns for both units are shown in Figures 4.3-1 and 4.3-2.

4.3.2.1.2 Fuel Bundle Nuclear Design

Reference 4.3-1 describes the first core bundle designs and related fuel nuclear properties. Reload fuel bundle design descriptions are presented below. The burnup dependent behavior of certain nuclear properties is primarily a function of enrichment and does not change significantly with bundle mechanical design. These characteristics include Uranium depletion and Plutonium buildup, fission fraction, delayed neutron fraction, and neutron lifetime. Figures 4.3-3 through 4.3-7 show the typical response of these characteristics with burnup for an enriched reload fuel bundle lattice.

ATRIUM-11 and ATRIUM-10 bundle designs are currently in use at SSES. Each design may have a unique axial enrichment distribution, radial enrichment distribution, or burnable absorber loading. Figures 4.3-8-46 and 4.3-8-47 show the nominal axial zoning for typical fuel bundles used in the reload cores. Figure series 4.3-9 shows the nominal radial enrichment distributions for typical lattice types used in the fuel bundles. The ATRIUM-11 fuel design utilizes chromium-doped UO_2 in non-burnable poison fuel pellets.

Unit/Cycle-specific reload fresh fuel bundle design descriptions may be found in References 4.3-19 and 4.3-20 for Unit 1 and Unit 2, respectively.

4.3.2.2 Power Distributions

This section presents typical power distributions for SSES reload cores. Typical local, core radial, and core axial power distributions for the initial core are described in Reference 4.3-1.

The core is designed such that the resulting power distributions meet the thermal limits identified in the plant Technical Specifications. The primary criteria for thermal limits are the Maximum Linear Heat Generation Rate (MLHGR) and the Minimum Critical Power Ratio (MCPR). In addition, a Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit is applied to the plant. Each of these parameters is a function of the core 3-D power distribution and the local rod-to-rod power distribution. Design calculations are performed to ensure that the core meets thermal limits and to demonstrate that the power distributions comply with the cycle design envelope.

The local peaking factor is defined as the ratio of the power density in the highest power rod in a lattice to the average power density in the lattice. Local effects on Critical Power Ratio are characterized by F-effective and the K-factor. The local peaking factor, F-effective, and the K-factor have associated target values which typically satisfy the design envelope. Gross power peaking in the core is defined as the ratio of the maximum power density in any axial segment of any bundle in the core to the average core power density. Design allowances are included in the design stage to ensure that thermal limits are met. During plant operation, the power distributions are measured by the in-core instrumentation system and thermal margins are calculated by the core monitoring system.

4.3.2.2.1 Local Power Distribution

The local rod-to-rod power distribution and associated F-effective and K-factor distributions are a direct function of the lattice enrichment distribution. Near the outside of the lattice where thermal flux peaks due to interbundle water gaps, low enrichment fuel rods are utilized to reduce power peaking. Closer to the center of the bundle, higher enrichment rods are used to increase power peaking and flatten the bundle power distribution. In addition, the water rods (or water channels) in the center of the lattice increase thermal flux and cause more power to be produced in the center of the lattice. The combination of enrichment and water channels results in a relatively flat power distribution.

To control bundle reactivity, Gd_2O_3 is utilized as a burnable absorber. Power is suppressed in gadolinia bearing fuel rods early in bundle life. As gadolinia is depleted, power in these rods initially increases, then decreases as fuel is depleted.
Local power distributions are calculated using licensed methodology described in Section 4.3.3.

Figures 4.3-11-1 and 4.3-11-8 show bundle reactivity (k_∞) as a function of void fraction and burnup for an ATRIUM-10 and an ATRIUM-11 fuel assembly dominant lattice, respectively. At low exposure, reactivity is higher for lower void fractions. As exposure increases the curves cross, largely due to the effect of void history and the increase in plutonium buildup.

Figures 4.3-11-2 to 4.3-11-4 and 4.3-11-9 to 4.3-11-11 show typical unrodded local power distributions for ATRIUM-10 and ATRIUM-11 fuel assembly dominant lattices as a function of burnup with a constant void fraction. Figures 4.3-11-2, 4.3-11-5, and 4.3-11-6 show typical unrodded local power distributions for a fresh ATRIUM-10 fuel dominant lattice as a function of void fraction at BOC. Figures 4.3-11-9, 4.3-11-12, and 4.3-11-13 show typical unrodded local power distributions for a fresh ATRIUM-11 fuel dominant lattice as a function of void fraction at BOC. Figures 4.3-11-7 and 4.3-11-14 show the typical response of the unrodded maximum local peaking factor as a function of void fraction and burnup for ATRIUM-10 and ATRIUM-11 fuel assemblies, respectively.

4.3.2.2.2 Radial Power Distribution

The integrated bundle power, commonly referred to as the radial power, is the primary factor for determining MCPR. At rated conditions the MCPR is directly proportional to the radial power. The radial power distribution is a function of the control rod pattern in the core, the fuel bundle type and loading pattern, and void distribution. Radial power is calculated using the licensed methodology described in Section 4.3.3.

4.3.2.2.3 Axial Power Distribution

Axial power distributions in a BWR are a function of control rod position, steam voids, axial gadolinia distribution, and the exposure distribution. Voids tend to skew power toward the bottom of the core; bottom entry control rods reduce the power in the bottom of the core; and the axial gadolinia distribution assists in flattening the power in the bottom of the core. Since the void distribution is primarily determined by the power shape, the two means available for axial power shape optimization are the control rods and gadolinia. Typically, the core axial power shape is bottom peaked at BOC and becomes top peaked at EOC.

Axial power shapes are calculated using the licensed methodology described in Section 4.3.3.

4.3.2.2.4 Power Distribution Measurements

Power distribution measurement methodology and measurement uncertainties are described in References 4.3-4, 4.3-10, and 4.3-13.

4.3.2.2.5 Power Distribution Accuracy

The accuracy of calculated power distributions is discussed in References 4.3-4, 4.3-10 and 4.3-13.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients are differential changes in reactivity produced by differential changes in core conditions. These coefficients are useful in calculating the response of the core to varying plant conditions. The initial condition of the core and the postulated initiating event determine which of the coefficients are significant in evaluating core response.

The dynamic behavior of BWRs over all operating states can be characterized by three reactivity coefficients. These coefficients are the Doppler coefficient, the moderator temperature coefficient, and the void coefficient. The Power coefficient is also associated with a BWR; however, this coefficient is the combination of the Doppler and void coefficients in the operating range.

Reactivity coefficients are calculated using the licensed methods described in Section 4.3.3.

4.3.2.3.1 Void Coefficient

The most important reactivity coefficient in a BWR is the void coefficient. The void coefficient must be large enough to prevent power oscillation due to spatial xenon changes, but it must be small enough that pressurization transients do not limit plant operation. The void coefficient inherently flattens the radial power distribution during normal operation and provides enhanced reactor control through the void feedback mechanism. The overall void coefficient is always negative over the complete operating range since the BWR design is typically undermoderated. Void formation changes reactivity by reducing the amount of water available for neutron moderation, thus increasing neutron leakage. Typical values for the void coefficient are listed in Table 4.3-4.

4.3.2.3.2 Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is the least important of the reactivity coefficients since its effect is limited to a very small portion of the reactor operating range. Once the reactor reaches the power production range, boiling begins and the MTC remains essentially constant. Like the void coefficient, the moderator coefficient is associated with the amount of neutron moderation in the water. The MTC is negative during power operation; however, under cold conditions beginning soon after BOC, the MTC may become slightly positive.

The range of values of MTCs in reload lattices does not include any that are significant from a safety point of view. Typical values for the MTC are listed in Table 4.3-4. The small magnitude of this coefficient, relative to that associated with steam voids, combined with the long time-constant associated with heat transfer from fuel to coolant, makes the reactivity contribution of a change in moderator temperature insignificant during rapid transients.

4.3.2.3.3 Doppler Temperature Coefficient

The Doppler Temperature coefficient (DTC) is the change in reactivity due to a change in fuel temperature. This change in reactivity occurs due to the broadening of the fuel resonance absorption cross sections as temperature increases.

The DTC is primarily a measure of the Doppler broadening of U238 and Pu240 resonance absorption peaks. An increase in fuel temperature increases the effective resonance absorption cross section of the fuel and produces a corresponding reduction in reactivity. The Doppler coefficient changes as a function of core life representing the combined effects of fuel temperature reduction with burnup and the buildup of Pu240. Typical values for the Doppler coefficient are listed in Table 4.3-4.

4.3.2.3.4 Power Coefficient

The power coefficient is determined from the composite of all the significant individual sources of reactivity change associated with a differential change in reactor power. This coefficient assumes constant xenon. Typical values for the power coefficient may be obtained from Reference 4.3-1 for the initial cores.

4.3.2.4 Control Requirements

The core and fuel design in conjunction with the reactivity control system provide a stable system for BWRs. The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the equilibrium fuel cycle operation. Since fuel reactivity is a maximum and control rod worth is a minimum at ambient temperature, the shutdown capability is evaluated assuming a cold, xenon free core. The safety design basis requires that the core, in its maximum reactivity condition, shall be subcritical with all control rods inserted except with the highest worth rod completely withdrawn. This limit allows control rod testing at any time in core life and assures that the reactor can be made subcritical by control rods alone.

The typical behavior of hot excess reactivity as a function of cycle exposure for SSES Units 1 and 2 is shown in Figure 4.3-12.

4.3.2.4.1 Shutdown Reactivity

Core Shutdown Margin calculations are performed to assess whether the basic criterion for reactivity control is met by the reload design. This criterion requires that the core, under cold, no xenon conditions, must be subcritical with the highest worth control rod fully withdrawn and all other rods fully inserted. SSES Technical Requirements Manual requires a shutdown margin of at least 0.38% $\Delta k/k$. The shutdown margin requirement is based on the uncertainties associated with the statistical variance of cold criticality calculations at a given exposure, plus a manufacturing uncertainty. The manufacturing uncertainty results from the fact that the calculated highest worth control rod may not be the highest worth rod in reality due to the stackup of manufacturing tolerances in a control cell.

Core Shutdown margin is very dependent on bundle and core designs and is a function of core exposure. Gadolinia loading, enrichment loading, and core loading all significantly affect core and local cell reactivity as a function of exposure. As a result, shutdown margin must be evaluated throughout the expected cycle operation to assure adequate margin to Technical Specification requirements. For design purposes, an additional uncertainty is added to the Technical Specification value to account for prediction uncertainties.

Shutdown margin is calculated as a function of cycle exposure in the following manner:

$$\mathrm{SDM}(E) = \frac{1-\left(k_{\mathrm{eff}}(E)-\mathrm{bias}(E)\right)}{k_{\mathrm{eff}}(E)-\mathrm{bias}(E)} * 100\%$$

where;

SDM(E) = core shutdown margin (%$\Delta k/k$) at cycle exposure E,

$K_{eff}(E)$ = core k-effective at cycle exposure (E) with all rods in except the strongest worth rod (no xenon, ≥ 68°F corresponding to the most reactive state),

Bias(E) = core k-effective bias for cold core simulation model at cycle exposure (E). The bias equals the target cold core simulation model critical k-effective minus 1.0.

The Cycle R value is determined from the evaluation of shutdown margin as a function of cycle exposure. The R value is used to determine shutdown margin testing requirements, and it is defined as the difference between the calculated beginning of cycle shutdown margin minus the calculated minimum shutdown margin during the cycle, where shutdown margin is a positive number. The value of R must be either positive or zero and must be determined for each fuel loading cycle.

Typical behavior of shutdown margin as a function of cycle exposure for SSES Units 1 and 2 is shown in Figure 4.3-13.

A description of the methods used to calculate shutdown margin is provided in Section 4.3.3.

4.3.2.4.2 Reactivity Variations

Reference 4.3-1 provides a general discussion of reactivity variations in a BWR/4. The reference provides tables showing typical k-effective values for various power levels, control fractions, and Xenon concentrations. From this data, the general reactivity effect of changing a single core variable can be determined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

4.3.2.5.1 RWM Range

Below the low power setpoint, control rod patterns follow prescribed withdrawal and insertion sequences restricted by the Rod Worth Minimizer (RWM). The sequences are established to assure that the maximum insequence control rod or rod segment reactivity worth would not be sufficient to result in a deposited fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal or insertion. Further discussion of the RWM and control rod sequence limitations is provided in Section 15.4.9 (Control Rod Drop Accident).

4.3.2.5.2 Operating Range

In the power range, above the low power setpoint, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak fuel enthalpy of 280 cal/gm. Therefore, restrictions on control rod patterns are not required to minimize control rod worths. During power operation the control rod patterns are selected based on the measured core power distributions.

For reload design purposes, optimized control rod patterns are selected for the cycle depletion. The series of design control rod patterns form the Cycle Step Out. Control rod sequence identification (A2, B2, A1, B1) is defined in Reference 4.3-1.

4.3.2.5.3 SCRAM Reactivity

The reactor protection system (RPS) is capable of shutting down the reactor by initiating a SCRAM. The RPS and the control rod drive (CRD) system act quickly enough to prevent the initiating event from driving the fuel beyond transient limits.

During a SCRAM from operating conditions, the control rod worth, reactor power, delayed neutron fraction, and void distributions must be properly accounted for as a function of time. The methodology used to account for these variables and determine SCRAM reactivity is described in Section 4.3.3.

4.3.2.6 Criticality of Reactor During Refueling

Criticality of fuel assemblies in the core during refueling is avoided by assuring that the Technical Specification shutdown margin requirement is met. For core shuffles, a shutdown margin design criterion is defined to account for prediction uncertainties. This criterion helps determine the acceptability of a fuel move for meeting the Technical Specification limit. A description of the methods used to calculate shutdown margin is provided in Section 4.3.3.

4.3.2.7 Stability

Boiling Water Reactors do not have instability problems due to Xenon. Xenon transients are highly damped in a BWR due to the large negative power coefficient. References 4.3-1 and 4.3-3 provide additional discussion of Xenon instability.

Thermal hydraulic stability is discussed in detail in Section 4.4.

4.3.2.8 Vessel Irradiations

The RAMA Fluence Methodology (Reference 4.3-14) is used to evaluate the Reactor Pressure Vessel (RPV) fluence for both units. This methodology has been reviewed and approved by the NRC for RPV fluence evaluations (Reference 4.3-15) and is consistent with applicable regulatory guidance (Reference 4.3-16). Detailed descriptions of the calculations for each unit are provided in References 4.3-7 and 4.3-8. The fast fluence evaluations are based on the RAMA Code Methodology. The Methodology includes a transport code, model builder codes, a fluence calculator code, an uncertainty methodology, and a nuclear data library. The transport code, fluence calculator, and nuclear data library are the primary software components for calculating the neutron flux and fluence. The transport code uses a deterministic, three-dimensional, multigroup nuclear particle transport theory to perform the neutron flux calculations. The transport code couples the nuclear transport method with a general geometry modeling capability to provide a flexible and accurate tool for calculating fluxes in light water reactors. The fluence calculator uses reactor operating history information with isotopic production and decay data to estimate activation and fluence in the reactor components over the operating life of the reactor. The nuclear data library contains nuclear cross-section data and response functions that are needed in the flux, fluence, and reaction rate calculations. The cross sections and response functions are based on the BUGLE-96 nuclear data library. Fluxes are calculated at the inner vessel surface, at ¼ T and ¾ T depths.

The RAMA methodology calculates RPV fluence and uncertainty at all locations in the RPV in the active core region in accordance with applicable regulatory guidance. The results from the vessel fluence coupon analyses are solely used to support the methodology uncertainty analysis. The RAMA methodology directly calculates the fluence at all RPV locations in the active core region. Therefore, lead factors, which were historically used to extrapolate the measured fluence at the coupon locations to the RPV ¼ T depths are no longer used or calculated.

Previous fluence calculations were performed using the DORT computer code, which is described in Section 4.1. The RAMA Fluence Methodology will continue to be used to calculate the fluence for both units and is described in BWRVIP-114 (Reference 4.3-14).

The analytical model for (R,θ) geometry is shown in Figure 4.3-14. The model consists of an inner and outer core region, the shroud, water regions inside and outside the shroud, jet pump components, the vessel wall, inner and outer Cavity, Mirror Insulation and the Biological Shield.

Neutron fluence was determined based on actual and expected operating history for each unit. This included the effects of several power uprates that have occurred during the operating history. Final end of life RPV fluence is calculated for both units at 54 EFPY at the RPV [both inner diameter (ID) and ¼ T (1/4 of the distance from the inside diameter to the outside diameter)] based on actual and expected operating history. Details on the power history assumed in the fluence analysis are provided in footnotes to the data in Table 4.3-5. Table 4.3-5 lists the 54 EFPY maximum fast fluence results and also provides historical results from the original analyses for comparison.

4.3.3 Analytical Methods

Reload design for SSES Units 1 and 2 is performed using NRC approved methodology. The approved methods used for nuclear design are fully described in Reference 4.3-13.

A summary description of several nuclear design codes is provided in Section 4.1.

Reference 4.3-1 describes the methods used for initial core nuclear design.

4.3.4 Changes

Reference 4.3-1 lists several changes made to the initial reactor nuclear design.

Reload core nuclear designs incorporate the following significant changes.

Unit 2, Cycle 9 and Unit 1, Cycle 11 were the first cores to utilize the ATRIUM-10 fuel design at SSES. ATRIUM-10 has a 10x10 lattice which is significantly different from the 9x9 lattice utilized in previous cycles. Nuclear characteristics of ATRIUM-10 fuel are discussed in Section 4.3. Mechanical design of ATRIUM-10 fuel is discussed in Section 4.2.

The CASMO-3G lattice physics code was first used to support the U1C10 reload design.

Unit 2, Cycle 9 was designed for a 24-month cycle. This cycle length represents a change from the 18-month cycle used for previous core designs. The effects of a 24-month cycle on the U2C9 reload were evaluated in Reference 4.3-11.

Unit 1, Cycle 11 was designed for a 24-month cycle. This cycle length represents a change from the 18-month cycle used for previous core designs. The effects of a 24-month cycle on the U1C11 reload were evaluated in Reference 4.3-12.

The CASMO-4/MICROBURN-B2 code system was first used to support the U1C14 reload design. A summary description of CASMO-4 and MICROBURN-B2 is provided in Section 4.1.

Unit 1 Cycle 14 was the first cycle to utilize 100 mil fuel channels and the Framatome-ANP FUELGUARD Lower Tie Plate design. The 100 mil fuel channel and FUELGUARD Lower Tie Plate are described in Section 4.2.

Unit 1 Cycle 20 was the first cycle to utilize the Advanced Fuel Channel (AFC). The AFC is described in Section 4.2.

Unit 2, Cycle 21 and Unit 1, Cycle 23 are the first cores to utilize the ATRIUM-11 fuel design at SSES. ATRIUM-11 has a 11x11 lattice which is different from the 10x10 lattice utilized in previous cycles. Nuclear characteristics of ATRIUM-11 fuel are discussed in Section 4.3. Mechanical design of ATRIUM-11 fuel is discussed in Section 4.2.

4.3.5 References

4.3-1 "BWR/4 and BWR/5 Fuel Design," NEDE-20944(P), General Electric Company, October 1976.

4.3-2 "BWR/4 and BWR/5 Fuel Design," Amendment 1 NEDE-20944-1(P), General Electric Company, January 1977.

4.3-3 Letter from Olan. D. Parr (NRC) to Dr. G. G. Sherwood (GE), "Review of General Electric Topical Report NEDE-20944-P, BWR/4 and BWR/5 Fuel Design (NEDO-20944 Non-Proprietary Version)," September 30, 1977.

4.3-4 "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," BAW-10247PA and Supplement 2P-A.

4.3-5 Deleted.

4.3-6 Deleted.

4.3-7 "Susquehanna Steam Electric Station Unit 1 Reactor Pressure Vessel Fluence Evaluation," SSE-FLU-001-R-002, Rev. 0, TransWare Enterprises, Inc., December 2019.

4.3-8 "Susquehanna Steam Electric Station Unit 2 Reactor Pressure Vessel Fluence Evaluation," SSE-FLU-001-R-001, Rev. 0, TransWare Enterprises, Inc., December 2019.

4.3-9 "Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2," NEDC-32161P, GE Nuclear Energy, December 1993.

4.3-10 "Advanced Nuclear Fuels Methodology For Boiling Water Reactors," XN-NF-80-19 (P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

4.3-11 "Susquehanna SES Unit 2 Cycle 9 Reload Summary Report," PL-NF-97-003, Rev. 1, PP&L, September 1997.

4.3-12 "Susquehanna SES Unit 1 Cycle 11 Reload Summary Report," PL-NF-98-002, Rev. 1, PP&L, Inc., July 1998.

4.3-13 EMF-2158 (P) (A), "Siemens Power Corporation Methodology For Boiling Water Reactors Evaluation and Validation of Casmo-4/Microburn-B2," October 1999.

4.3-14 "BWR vessel and Internals Project RAMA Fluence Methodology Theory Manual," BWRVIP-114, June 2009.

4.3-15 Safety Evaluation of proprietary EPRI Reports, “BWR Vessel And Internals Project, RAMA Fluence Methodology Manual (BWRVIP-114),” “RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115),” “RAMA Fluence Methodology - Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117),” and “RAMA Fluence Methodology Procedures Manual (BWRVIP-121),” and “Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)” (TAC No. MB9765), BWRVIP 2005-208B, William H. Bateman, NRC to Bill Eaton, BWRVIP Chairman, May 13, 2005.

4.3-16 “Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” Regulatory Guide 1.190, March 2001.

4.3-17 “Susquehanna Unit 1 Cycle 23 Fuel Cycle Design Report,” ANP-3953P, Rev. 0, Framatome Inc., August 2021, (General Reference per NEI 98-03).

4.3-18 “Susquehanna Unit 2 Cycle 22 Fuel Cycle Design Report,” ANP-4008P, Rev. 0, Framatome Inc., August 2022, (General Reference per NEI 98-03).

4.3-19 “Susquehanna Unit 1 Cycle 23 ATRIUM 11 Fuel Nuclear Fuel Design Report,” ANP-3951P, Rev. 0, Framatome Inc., August 2021, (General Reference per NEI 98-03).

4.3-20 “Susquehanna Unit 2 Cycle 22 ATRIUM-11 Fuel Nuclear Fuel Design Report,” ANP-4007P, Rev. 0., Framatome Inc., July 2022, (General Reference per NEI 98-03).

TABLE 4.3-1

REACTOR CORE CHARACTERISTICS

Reactor Type/Configuration	BWR-4/2 Loop Jet Pump Recirculation System, C-Lattice
Rated Thermal Power, Unit 1	3952 MWt
Rated Thermal Power, Unit 2	3952 Mwt
Number of Fuel Assemblies	764
Number of Control Rods	185
Number of Traversing In-core Probe Locations	43
Active Core Height, ft	12.45
Control Rod Pitch, inches	12.0
Fuel Assembly Pitch, inches	6.0

Table 4.3-4
TYPICAL CORE REACTIVITY COEFFICIENTS

Security-Related Information
Table Withheld Under 10 CFR 2.390

Table 4.3-5
FAST NEUTRON FLUENCES > 1 Mev
(54 EFPY Fluence)[1]

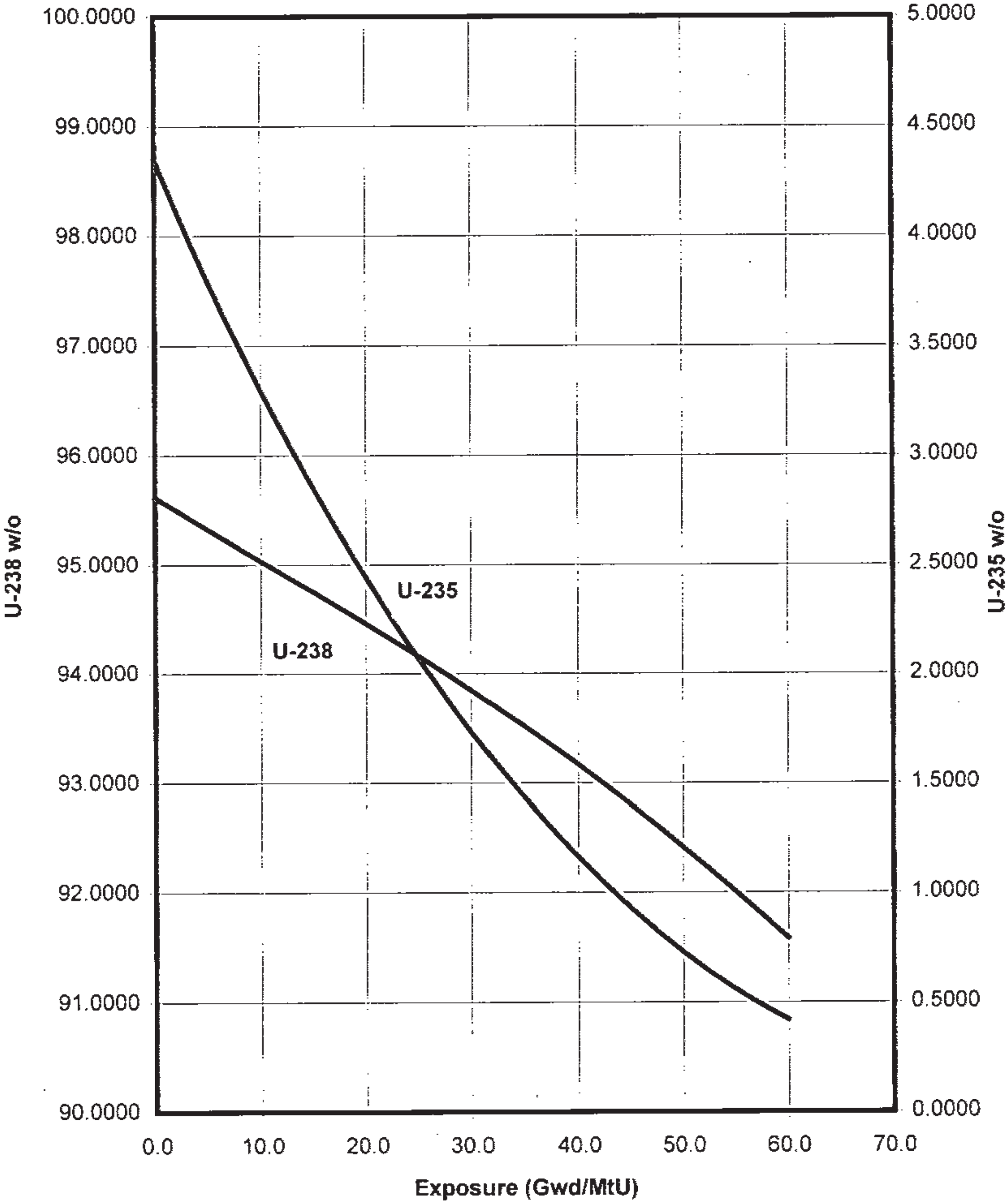
Security-Related Information
Table Withheld Under 10 CFR 2.390

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE LOADING MAP TYPICAL OF UNIT 1 FIGURE 4.3-1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE LOADING MAP TYPICAL OF UNIT 2 FIGURE 4.3-2



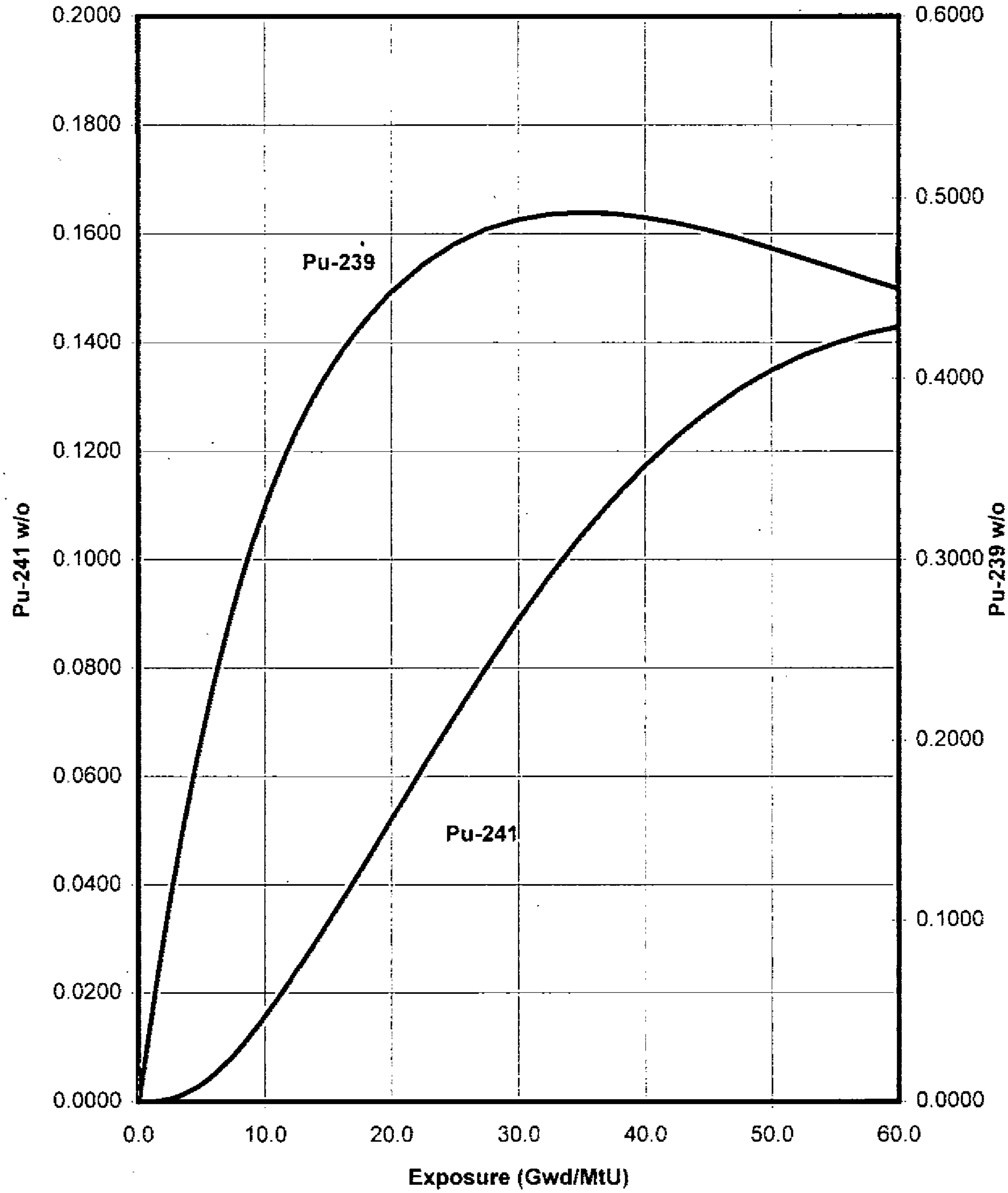
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

URANIUM DEPLETION AS A FUNCTION OF
EXPOSURE, 40% VOIDS
(TYPICAL)

FIGURE 4.3-3, Rev. 54

Auto Cad: Figure Fsar 4_3_3.dwg



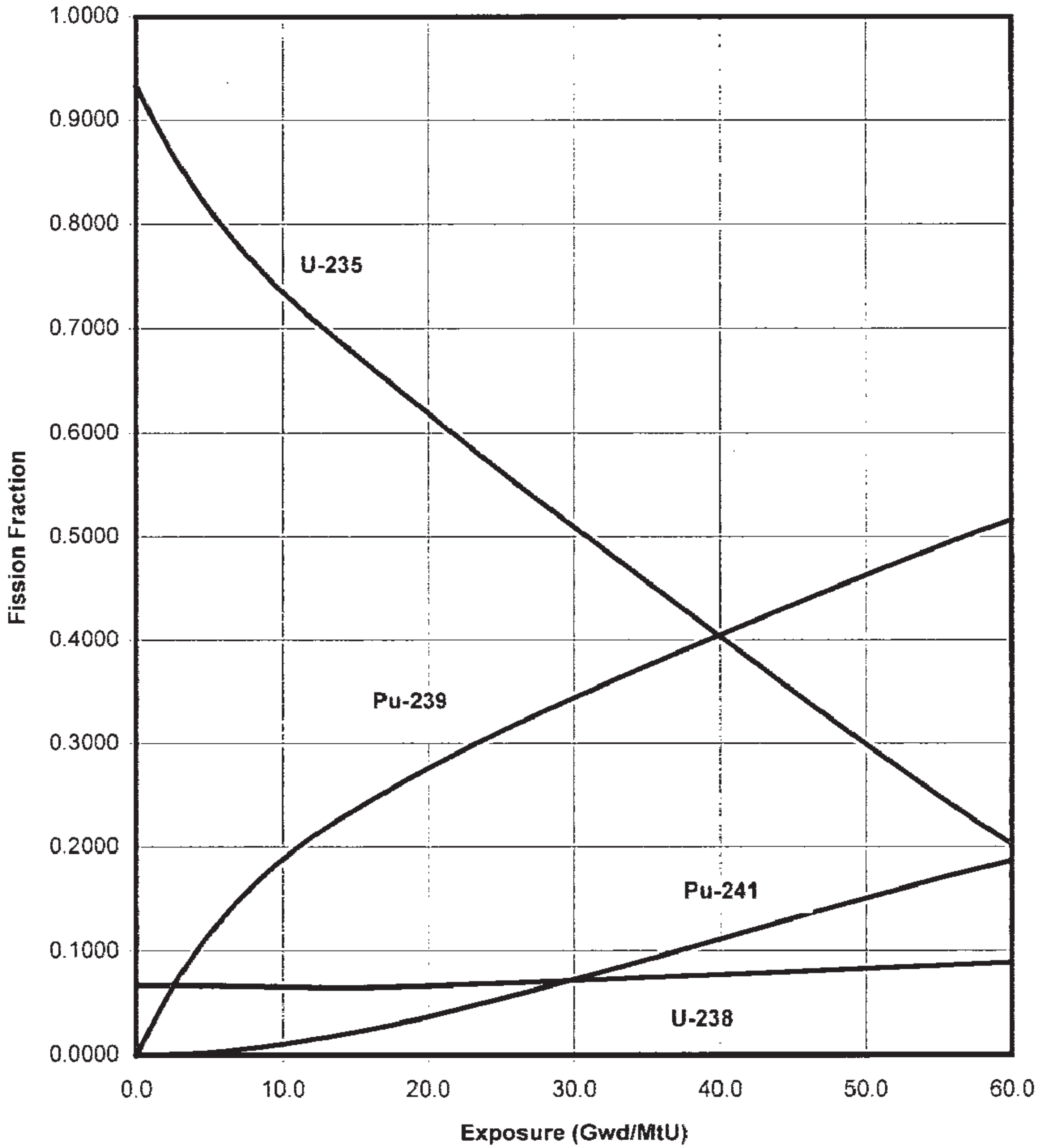
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PLUTONIUM BUILDUP AS A FUNCTION OF
EXPOSURE, 40% VOIDS
(TYPICAL)

FIGURE 4.3-4, Rev. 54

Auto Cad: Figure Fsar 4_3_4.dwg



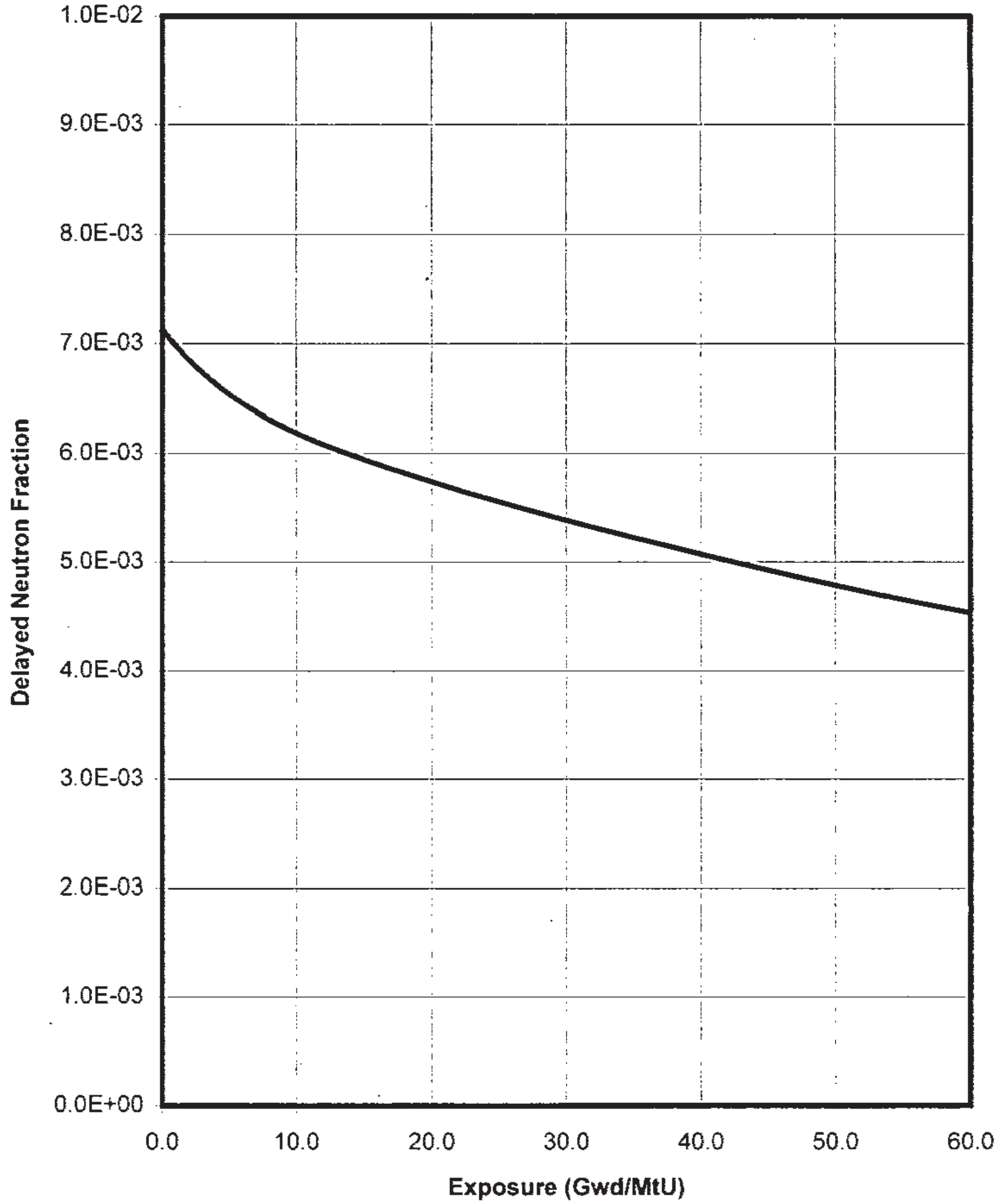
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FISSION FRACTION AS A FUNCTION OF
EXPOSURE, 40% VOIDS
(TYPICAL)

FIGURE 4.3-5, Rev. 54

Auto Cad: Figure Fsar 4_3_5.dwg



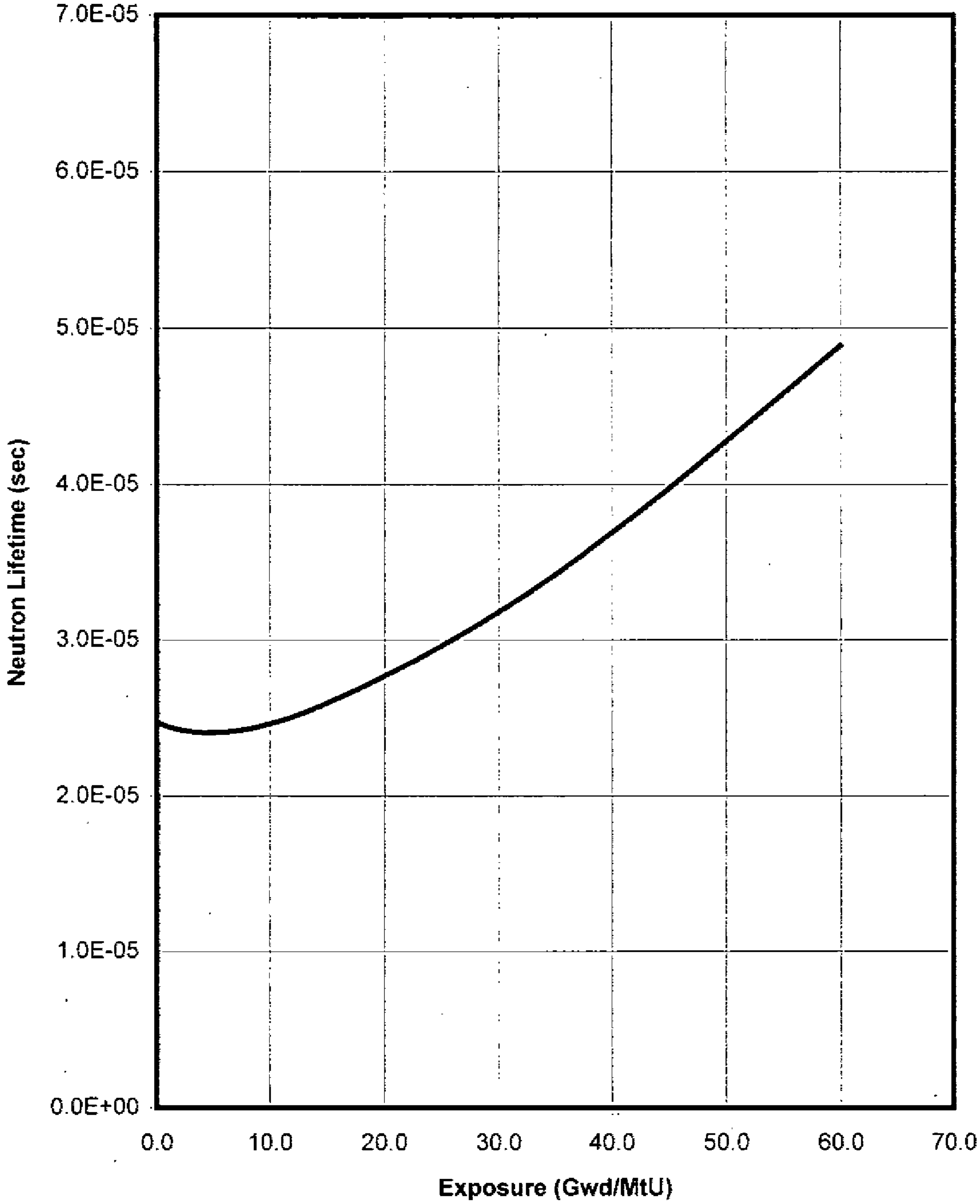
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

DELAYED NEUTRON FRACTION AS A
FUNCTION OF EXPOSURE
40% VOIDS
(TYPICAL)

FIGURE 4.3-6, Rev. 54

Auto Cad: Figure Fsar 4_3_6.dwg



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SUSQUEHANNA STEAM ELECTRIC STATION
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NEUTRON LIFETIME AS A FUNCTION OF EXPOSURE, 40% VOIDS (TYPICAL)

FIGURE 4.3-7, Rev. 54

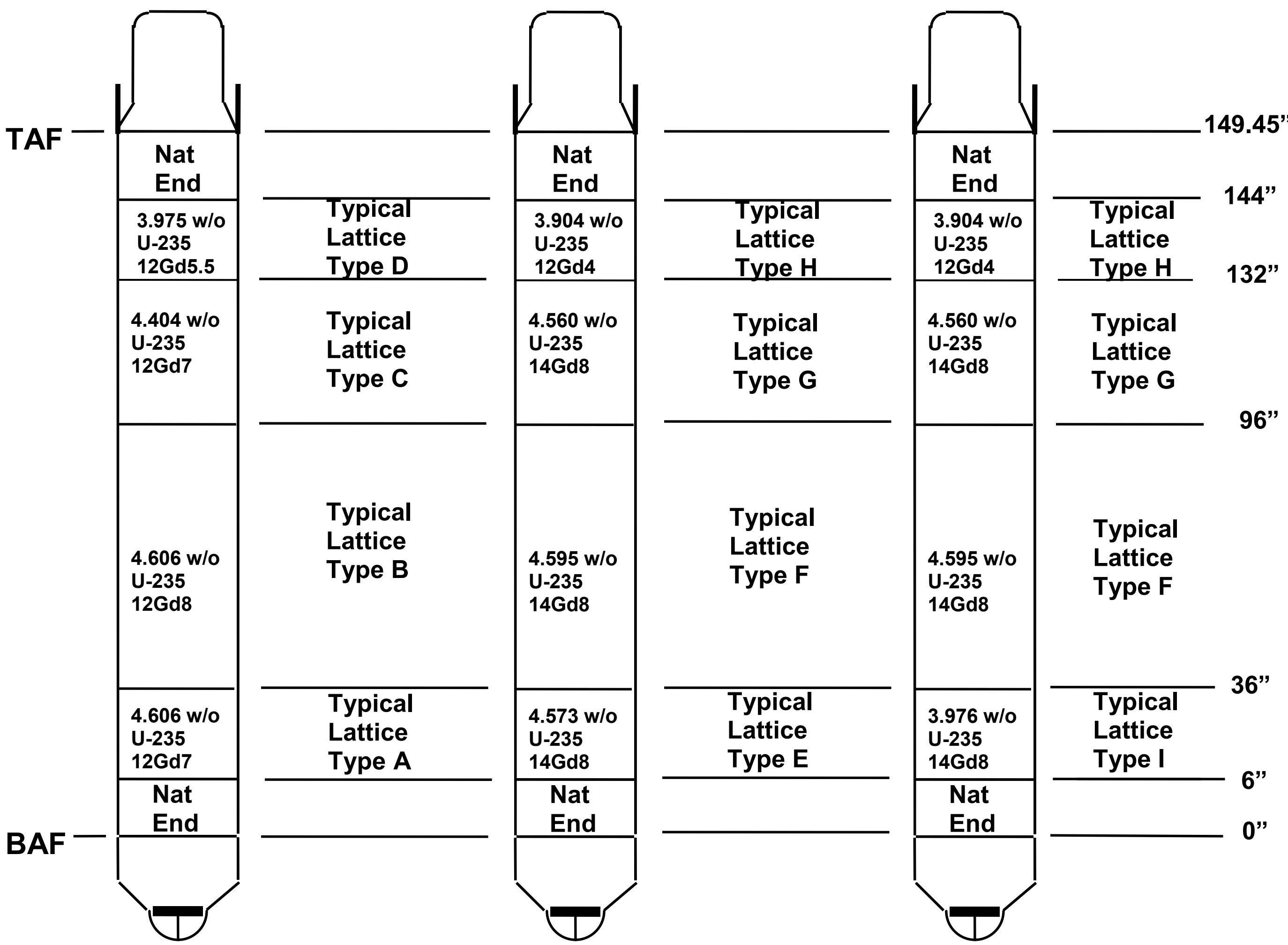
Auto Cad: Figure Fsar 4_3_7.dwg

Typical Assembly Types

Reload Bundle Description

(ATRIUM-10 with Standard or Advanced Fuel Channel)

3.7% < Bundle Average Enrichment < 4.3%



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SUSQUEHANNA STEAM ELECTRIC STATION
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FINAL SAFETY ANALYSIS REPORT

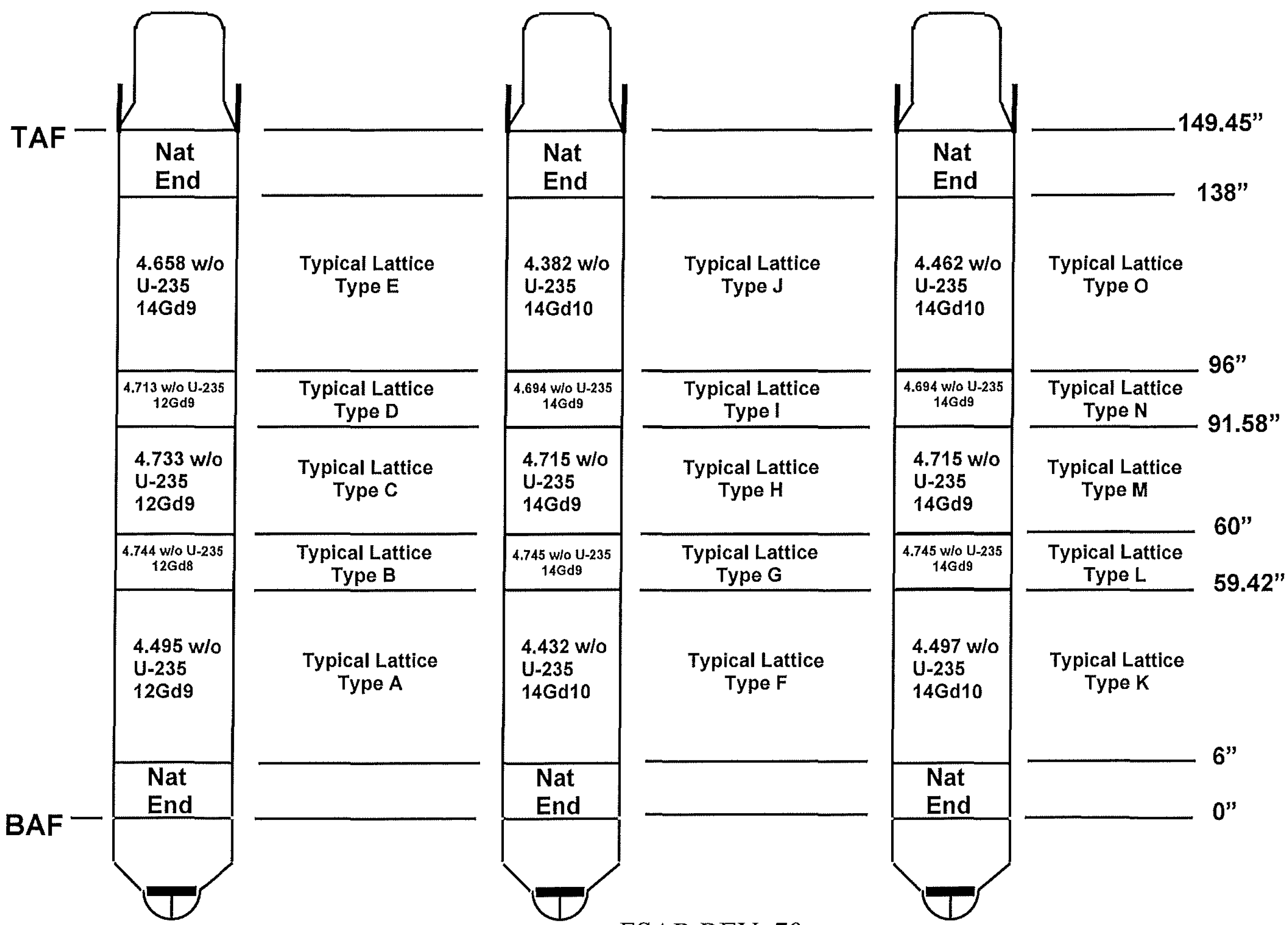
ATRIUM™-10 FUEL AXIAL ENRICHMENT
(NOMINAL) TYPICAL ASSEMBLIES

FIGURE 4.3-8-46, Rev 0

Typical Assembly Types

Reload Bundle Description

(ATRIUM 11 with Advanced Fuel Channel)



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ATRIUM 11 FUEL AXIAL ENRICHMENT
(NOMINAL) TYPICAL ASSEMBLIES

FIGURE 4.3-8-47, Rev 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95				E5 4.95	G1 4.45 7.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 7.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 7.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 7.00	E5 4.95	G1 4.45 7.00	E5 4.95	G1 4.45 7.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	12	4.45 + 7.00
E4	2	4.70
E5	57	4.95

A10B-4606L-12G70-FS10 Enrichment Distribution

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE A

Figure 4.3-9-123, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	12	4.45 + 8.00
E4	2	4.70
E5	57	4.95

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE B

Figure 4.3-9-124, Rev. 0

E1 2.40	E2 3.20	E3 4.00	E4 4.45	E4 4.45	E4 4.45	E4 4.45	E3 4.00	E2 3.20	E1 2.40
E2 3.20		G1 4.45 7.00	E6 4.95		E6 4.95	G1 4.45 7.00	E6 4.95		E2 3.20
E3 4.00	G1 4.45 7.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	G1 4.45 7.00	E3 4.00
E4 4.45	E6 4.95	E6 4.95	G1 4.45 7.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E5 4.70
E4 4.45		E6 4.95	E6 4.95				E6 4.95	G1 4.45 7.00	E5 4.70
E4 4.45	E6 4.95	E6 4.95	E6 4.95		Water Channel		E5 4.70		E5 4.70
E4 4.45	G1 4.45 7.00	E6 4.95	E6 4.95				E4 4.45	G1 4.45 7.00	E5 4.70
E3 4.00	E6 4.95	E6 4.95	E6 4.95	E6 4.95	E5 4.70	E4 4.45	G1 4.45 7.00	E6 4.95	E3 4.00
E2 3.20		G1 4.45 7.00	E6 4.95	G1 4.45 7.00		G1 4.45 7.00	E6 4.95		E2 3.20
E1 2.40	E2 3.20	E3 4.00	E5 4.70	E5 4.70	E5 4.70	E5 4.70	E3 4.00	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	4.00
E4	10	4.45
G1	12	4.45 + 7.00
E5	10	4.70
E6	31	4.95

A10T-4404L-12G70-FS10 Enrichment Distribution

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UNITS 1 AND 2
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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE C

Figure 4.3-9-125, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20		G1 4.00 5.50	E5 4.45		E5 4.45	G1 4.00 5.50	E5 4.45		E2 3.20
E3 3.60	G1 4.00 5.50	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	G1 4.00 5.50	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 4.00 5.50	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00
E4 4.00		E5 4.45	E5 4.45				E5 4.45	G1 4.00 5.50	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45		Water Channel		E4 4.00		E4 4.00
E4 4.00	G1 4.00 5.50	E5 4.45	E5 4.45				E4 4.00	G1 4.00 5.50	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	G1 4.00 5.50	E5 4.45	E3 3.60
E2 3.20		G1 4.00 5.50	E5 4.45	G1 4.00 5.50		G1 4.00 5.50	E5 4.45		E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
E4	20	4.00
G1	12	4.00 + 5.50
E5	31	4.45

A10T-3975L-12G55-FS10 Enrichment Distribution

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE D

Figure 4.3-9-126, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E4 4.70
E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E4 4.70
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E3 4.45
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E4 4.70	E4 4.70	E4 4.70	E3 4.45	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	10	4.45
G1	14	4.45 + 8.00
E4	6	4.70
E5	49	4.95

A10B-4573L-14G80-FS10 Enrichment Distribution

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE E

Figure 4.3-9-127, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	14	4.45 + 8.00
E4	2	4.70
E5	55	4.95

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE F

Figure 4.3-9-128, Rev. 0

E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.45	E2 3.60	E1 2.40
E2 3.60		G1 4.45 8.00	E5 4.95		E5 4.95	G1 4.45 8.00	E5 4.95		E2 3.60
E3 4.45	G1 4.45 8.00	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E3 4.45
E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95
E5 4.95		G1 4.45 8.00	E5 4.95				E5 4.95	G1 4.45 8.00	E5 4.95
E5 4.95	E5 4.95	E5 4.95	E5 4.95	Water Channel			E5 4.95		E5 4.95
E5 4.95	G1 4.45 8.00	E5 4.95	E5 4.95				E5 4.95	G1 4.45 8.00	E4 4.70
E3 4.45	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.45 8.00	E5 4.95	E3 4.45
E2 3.60		G1 4.45 8.00	E5 4.95	G1 4.45 8.00		G1 4.45 8.00	E5 4.95		E2 3.60
E1 2.40	E2 3.60	E3 4.45	E5 4.95	E5 4.95	E5 4.95	E4 4.70	E3 4.45	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	8	4.45
G1	14	4.45 + 8.00
E4	2	4.70
E5	47	4.95

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(NOMINAL)
TYPICAL LATTICE G

Figure 4.3-9-129, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20		G1 3.80 4.00	E5 4.45		E5 4.45	G1 3.80 4.00	E5 4.45		E2 3.20
E3 3.60	G1 3.80 4.00	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	G1 3.80 4.00	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 3.80 4.00	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	E4 4.00
E4 4.00		E5 4.45	E5 4.45				E4 4.00	G1 3.80 4.00	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45		Water Channel		E4 4.00		E4 4.00
E4 4.00	G1 3.80 4.00	E5 4.45	E5 4.45				E4 4.00	G1 3.80 4.00	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E4 4.00	E4 4.00	E4 4.00	E4 4.00	G1 3.80 4.00	E4 4.00	E3 3.60
E2 3.20		G1 3.80 4.00	E4 4.00	G1 3.80 4.00		G1 3.80 4.00	E4 4.00		E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
G1	12	3.80 + 4.00
E4	28	4.00
E5	23	4.45

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE H

Figure 4.3-9-130, Rev. 0

E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40
E2 3.20	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E2 3.20
E3 3.60	G1 3.80 8.00	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	G1 3.80 8.00	E3 3.60
E4 4.00	E5 4.45	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00
E4 4.00	E5 4.45	G1 3.80 8.00	E5 4.45				E5 4.45	G1 3.80 8.00	E4 4.00
E4 4.00	E5 4.45	E5 4.45	E5 4.45	Water Channel			E4 4.00	E5 4.45	E4 4.00
E4 4.00	G1 3.80 8.00	E5 4.45	E5 4.45				E4 4.00	G1 3.80 8.00	E4 4.00
E3 3.60	E5 4.45	E5 4.45	E5 4.45	E5 4.45	E4 4.00	E4 4.00	G1 3.80 8.00	E5 4.45	E3 3.60
E2 3.20	E5 4.45	G1 3.80 8.00	E5 4.45	G1 3.80 8.00	E5 4.45	G1 3.80 8.00	E5 4.45	E5 4.45	E2 3.20
E1 2.40	E2 3.20	E3 3.60	E4 4.00	E4 4.00	E4 4.00	E4 4.00	E3 3.60	E2 3.20	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.20
E3	8	3.60
G1	14	3.80 + 8.00
E4	20	4.00
E5	37	4.45

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ATRIUM™-10 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE I

Figure 4.3-9-131, Rev. 0

E1 2.40	E2 3.60	E3 4.00	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E3 4.00	E2 3.60	E1 2.40
E2 3.60	G2 4.40 9.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.00 6.00	E5 4.95	E3 4.00	E2 3.60
E3 4.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.00
E4 4.40	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G1 4.00 6.00	E3 4.00
E4 4.40	E5 4.95	E5 4.95	E5 4.95				G2 4.40 9.00	E5 4.95	E5 4.95	E4 4.40
E4 4.40	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95	E4 4.40
E4 4.40	E5 4.95	E5 4.95	E5 4.95				G2 4.40 9.00	E5 4.95	E5 4.95	E4 4.40
E4 4.40	G1 4.00 6.00	E5 4.95	E5 4.95	G2 4.40 9.00	E5 4.95	G2 4.40 9.00	E5 4.95	E5 4.95	G1 4.00 6.00	E4 4.40
E3 4.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.00
E2 3.60	E3 4.00	E5 4.95	G1 4.00 6.00	E5 4.95	E5 4.95	E5 4.95	G1 4.00 6.00	E5 4.95	G2 4.40 9.00	E2 3.60
E1 2.40	E2 3.60	E3 4.00	E3 4.00	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E3 4.00	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	12	4.00
G1	6	4.00 + 6.00
E4	18	4.40
G2	6	4.40 + 9.00
E5	58	4.95

A11B-4495L-12GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE A

Figure 4.3-9-132, Rev. 0

EVC	E1 3.60	E3 4.70	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.70	E1 3.60	EVC
E1 3.60	G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	E2 4.40	E1 3.60
E3 4.70	E4 4.95	EVC	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	EVC	E4 4.95	E3 4.70
E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	EVC	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E3 4.70
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	EVC		Water Channel		EVC	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	G1 4.40 8.00	E4 4.95	E4 4.95	G1 4.40 8.00	EVC	G1 4.40 8.00	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95
E3 4.70	E4 4.95	EVC	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	EVC	E4 4.95	E3 4.70
E1 3.60	E2 4.40	E4 4.95	G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	G1 4.40 8.00	E1 3.60
EVC	E1 3.60	E3 4.70	E3 4.70	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.70	E1 3.60	EVC

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	2	4.40
G1	12	4.40 + 8.00
E3	10	4.70
E4	68	4.95

A11MP-4744L-12G80 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE B

Figure 4.3-9-133, Rev. 0

PLF	E1 3.60	E2 4.40	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	E2 4.40	E1 3.60
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	PLF	E4 4.95	E3 4.70
E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E3 4.70
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	PLF		Water Channel		PLF	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	G1 4.40 8.00	E4 4.95	E4 4.95	G2 4.40 9.00	PLF	G2 4.40 9.00	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	PLF	E4 4.95	E3 4.70
E1 3.60	E2 4.40	E4 4.95	G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	G2 4.40 9.00	E1 3.60
PLF	E1 3.60	E3 4.70	E3 4.70	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.70	E1 3.60	PLF

Type U-235

Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	6	4.40
G1	6	4.40 + 8.00
G2	6	4.40 + 9.00
E3	6	4.70
E4	68	4.95

A11M-4733L-12GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE C

Figure 4.3-9-134, Rev. 0

PLF	E1 3.60	E2 4.40	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	E2 4.40	E1 3.60
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	EVC	E4 4.95	E4 4.95	PLF	E4 4.95	E3 4.70
E4 4.95	E4 4.95	E4 4.95	EVC	E4 4.95	PLF	E4 4.95	EVC	E4 4.95	G1 4.40 8.00	E3 4.70
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	EVC	PLF		Water Channel		PLF	EVC	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	E4 4.95				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	G1 4.40 8.00	E4 4.95	EVC	G2 4.40 9.00	PLF	G2 4.40 9.00	EVC	E4 4.95	G1 4.40 8.00	E4 4.95
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	EVC	E4 4.95	E4 4.95	PLF	E4 4.95	E3 4.70
E1 3.60	E2 4.40	E4 4.95	G1 4.40 8.00	E4 4.95	E4 4.95	E4 4.95	G1 4.40 8.00	E4 4.95	G2 4.40 9.00	E1 3.60
PLF	E1 3.60	E3 4.70	E3 4.70	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.70	E1 3.60	PLF

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	6	4.40
G1	6	4.40 + 8.00
G2	6	4.40 + 9.00
E3	6	4.70
E4	60	4.95

A11TP-4713L-12GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE D

Figure 4.3-9-135, Rev. 0

PLF	E1 3.60	E2 4.40	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G1 4.00 6.00	E4 4.95	G2 4.40 9.00	E1 3.60
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.40
E4 4.95	E4 4.95	E4 4.95	PLF	E3 4.70	PLF	E3 4.70	PLF	E4 4.95	G1 4.00 6.00	E2 4.40
E4 4.95	E4 4.95	E4 4.95	E3 4.70				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	E4 4.95	PLF	PLF		Water Channel		PLF	PLF	E4 4.95	E4 4.95
E4 4.95	E4 4.95	E4 4.95	E3 4.70				G2 4.40 9.00	E4 4.95	E4 4.95	E4 4.95
E4 4.95	G1 4.00 6.00	E4 4.95	PLF	G2 4.40 9.00	PLF	G2 4.40 9.00	PLF	E4 4.95	G1 4.00 6.00	E4 4.95
E2 4.40	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.40
E1 3.60	G2 4.40 9.00	E4 4.95	G1 4.00 6.00	E4 4.95	E4 4.95	E4 4.95	G1 4.00 6.00	E4 4.95	G2 4.40 9.00	E1 3.60
PLF	E1 3.60	E2 4.40	E2 4.40	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E2 4.40	E1 3.60	PLF

Type
U-235

Type
U-235
Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
G1	6	4.00 + 6.00
E2	10	4.40
G2	8	4.40 + 9.00
E3	4	4.70
E4	56	4.95

A11T-4658L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE E

Figure 4.3-9-136, Rev. 0

E1 2.40	E2 3.60	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E2 3.60	E1 2.40
E2 3.60	G2 4.00 10.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G3 4.40 8.00	E4 4.95	G1 4.00 9.00	E2 3.60
E3 4.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.00
E3 4.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G3 4.40 8.00	E3 4.00
E3 4.00	E4 4.95	E4 4.95	E4 4.95				G2 4.00 10.00	E4 4.95	E4 4.95	E3 4.00
E3 4.00	E4 4.95	E4 4.95	E4 4.95		Water Channel		E4 4.95	E4 4.95	E4 4.95	E3 4.00
E3 4.00	E4 4.95	E4 4.95	E4 4.95				G2 4.00 10.00	E4 4.95	E4 4.95	E3 4.00
E3 4.00	G3 4.40 8.00	E4 4.95	E4 4.95	G2 4.00 10.00	E4 4.95	G2 4.00 10.00	E4 4.95	E4 4.95	G3 4.40 8.00	E3 4.00
E3 4.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E3 4.00
E2 3.60	G1 4.00 9.00	E4 4.95	G3 4.40 8.00	E4 4.95	E4 4.95	E4 4.95	G3 4.40 8.00	E4 4.95	G2 4.00 10.00	E2 3.60
E1 2.40	E2 3.60	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E3 4.00	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	28	4.00
G1	2	4.00 + 9.00
G2	6	4.00 + 10.00
G3	6	4.40 + 8.00
E4	58	4.95

A11B-4432L-14GV100 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT (NOMINAL)
TYPICAL LATTICE F

Figure 4.3-9-137, Rev. 0

EVC	E1 3.60	E2 4.70	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.70	E1 3.60	EVC
E1 3.60	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
E2 4.70	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E2 4.70
E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E2 4.70
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	EVC		Water Channel		EVC	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	G2 4.40 9.00	EVC	G2 4.40 9.00	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95
E2 4.70	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E2 4.70
E1 3.60	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
EVC	E1 3.60	E2 4.70	E2 4.70	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.70	E1 3.60	EVC

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E2	10	4.70
E3	68	4.95

A11MP-4745L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE G

Figure 4.3-9-138, Rev. 0

PLF	E1 3.60	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E2 4.40
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	PLF		Water Channel		PLF	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	G2 4.40 9.00	PLF	G2 4.40 9.00	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E1 3.60	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
PLF	E1 3.60	E2 4.40	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	10	4.40
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E3	68	4.95

A11M-4715L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE H

Figure 4.3-9-139, Rev. 0

<table>
<tr><td>PLF</td><td>E1
3.60</td><td>E2
4.40</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E2
4.40</td><td>E1
3.60</td><td>PLF</td></tr>
<tr><td>E1
3.60</td><td>G1
4.40
8.00</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
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4.40
8.00</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E1
3.60</td></tr>
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4.95</td><td>E3
4.95</td><td>EVC</td><td>E3
4.95</td><td>E3
4.95</td><td>PLF</td><td>E3
4.95</td><td>E2
4.40</td></tr>
<tr><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>EVC</td><td>E3
4.95</td><td>PLF</td><td>E3
4.95</td><td>EVC</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E2
4.40</td></tr>
<tr><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td rowspan="3" colspan="3">Water Channel</td><td>G2
4.40
9.00</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td></tr>
<tr><td>E3
4.95</td><td>E3
4.95</td><td>EVC</td><td>PLF</td><td>PLF</td><td>EVC</td><td>E3
4.95</td><td>E3
4.95</td></tr>
<tr><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>G2
4.40
9.00</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td></tr>
<tr><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E3
4.95</td><td>EVC</td><td>G2
4.40
9.00</td><td>PLF</td><td>G2
4.40
9.00</td><td>EVC</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E3
4.95</td></tr>
<tr><td>E2
4.40</td><td>E3
4.95</td><td>PLF</td><td>E3
4.95</td><td>E3
4.95</td><td>EVC</td><td>E3
4.95</td><td>E3
4.95</td><td>PLF</td><td>E3
4.95</td><td>E2
4.40</td></tr>
<tr><td>E1
3.60</td><td>G1
4.40
8.00</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E3
4.95</td><td>G1
4.40
8.00</td><td>E1
3.60</td></tr>
<tr><td>PLF</td><td>E1
3.60</td><td>E2
4.40</td><td>E2
4.40</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E3
4.95</td><td>E2
4.40</td><td>E1
3.60</td><td>PLF</td></tr>
</table>

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	10	4.40
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E3	60	4.95

A11TP-4694L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE I

Figure 4.3-9-140, Rev. 0

PLF	E1 3.60	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E1 3.60	PLF
E1 3.60	G2 4.00 10.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G1 4.00 9.00	E4 4.95	G1 4.00 9.00	E1 3.60
E2 4.00	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.00
E2 4.00	E4 4.95	E4 4.95	PLF	E3 4.40	PLF	E3 4.40	PLF	E4 4.95	G1 4.00 9.00	E2 4.00
E2 4.00	E4 4.95	E4 4.95	E3 4.40				G2 4.00 10.00	E4 4.95	E4 4.95	E2 4.00
E2 4.00	E4 4.95	PLF	PLF		Water Channel		PLF	PLF	E4 4.95	E2 4.00
E2 4.00	E4 4.95	E4 4.95	E3 4.40				G2 4.00 10.00	E4 4.95	E4 4.95	E2 4.00
E2 4.00	G1 4.00 9.00	E4 4.95	PLF	G2 4.00 10.00	PLF	G2 4.00 10.00	PLF	E4 4.95	G1 4.00 9.00	E2 4.00
E2 4.00	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.00
E1 3.60	G1 4.00 9.00	E4 4.95	G1 4.00 9.00	E4 4.95	E4 4.95	E4 4.95	G1 4.00 9.00	E4 4.95	G2 4.00 10.00	E1 3.60
PLF	E1 3.60	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E2 4.00	E1 3.60	PLF

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	28	4.00
G1	8	4.00 + 9.00
G2	6	4.00 + 10.00
E3	4	4.40
E4	38	4.95

A11T-4382L-14GV100 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE J

Figure 4.3-9-141, Rev. 0

E1 2.40	E2 3.60	E3 4.00	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E3 4.00	E2 3.60	E1 2.40
E2 3.60	G2 4.00 10.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G3 4.40 8.00	E5 4.95	G1 4.00 6.00	E2 3.60
E3 4.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.00
E4 4.40	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	G3 4.40 8.00	E3 4.00
E4 4.40	E5 4.95	E5 4.95	E5 4.95				G2 4.00 10.00	E5 4.95	E5 4.95	E4 4.40
E4 4.40	E5 4.95	E5 4.95	E5 4.95		Water Channel		E5 4.95	E5 4.95	E5 4.95	E4 4.40
E4 4.40	E5 4.95	E5 4.95	E5 4.95				G2 4.00 10.00	E5 4.95	E5 4.95	E4 4.40
E4 4.40	G3 4.40 8.00	E5 4.95	E5 4.95	G2 4.00 10.00	E5 4.95	G2 4.00 10.00	E5 4.95	E5 4.95	G3 4.40 8.00	E4 4.40
E3 4.00	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E5 4.95	E3 4.00
E2 3.60	G1 4.00 6.00	E5 4.95	G3 4.40 8.00	E5 4.95	E5 4.95	E5 4.95	G3 4.40 8.00	E5 4.95	G2 4.00 10.00	E2 3.60
E1 2.40	E2 3.60	E3 4.00	E3 4.00	E4 4.40	E4 4.40	E4 4.40	E4 4.40	E3 4.00	E2 3.60	E1 2.40

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	4	2.40
E2	8	3.60
E3	10	4.00
G1	2	4.00 + 6.00
G2	6	4.00 + 10.00
E4	18	4.40
G3	6	4.40 + 8.00
E5	58	4.95

A11B-4497L-14GV100 Enrichment Distribution

FSAR REV. 70

**SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT**

**ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE K**

Figure 4.3-9-142, Rev. 0

EVC	E1 3.60	E2 4.70	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.70	E1 3.60	EVC
E1 3.60	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
E2 4.70	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E2 4.70
E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E2 4.70
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	EVC		Water Channel		EVC	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	G2 4.40 9.00	EVC	G2 4.40 9.00	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95
E2 4.70	E3 4.95	EVC	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	E2 4.70
E1 3.60	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
EVC	E1 3.60	E2 4.70	E2 4.70	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.70	E1 3.60	EVC

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E2	10	4.70
E3	68	4.95

A11MP-4745L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE L

Figure 4.3-9-143, Rev. 0

PLF	E1 3.60	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E2 4.40
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	PLF		Water Channel		PLF	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	G2 4.40 9.00	PLF	G2 4.40 9.00	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E1 3.60	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
PLF	E1 3.60	E2 4.40	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	10	4.40
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E3	68	4.95

A11M-4715L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE M

Figure 4.3-9-144, Rev. 0

PLF	E1 3.60	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF
E1 3.60	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	EVC	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E3 4.95	E3 4.95	E3 4.95	EVC	E3 4.95	PLF	E3 4.95	EVC	E3 4.95	G1 4.40 8.00	E2 4.40
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	E3 4.95	EVC	PLF		Water Channel		PLF	EVC	E3 4.95	E3 4.95
E3 4.95	E3 4.95	E3 4.95	E3 4.95				G2 4.40 9.00	E3 4.95	E3 4.95	E3 4.95
E3 4.95	G1 4.40 8.00	E3 4.95	EVC	G2 4.40 9.00	PLF	G2 4.40 9.00	EVC	E3 4.95	G1 4.40 8.00	E3 4.95
E2 4.40	E3 4.95	PLF	E3 4.95	E3 4.95	EVC	E3 4.95	E3 4.95	PLF	E3 4.95	E2 4.40
E1 3.60	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E3 4.95	E3 4.95	E3 4.95	G1 4.40 8.00	E3 4.95	G1 4.40 8.00	E1 3.60
PLF	E1 3.60	E2 4.40	E2 4.40	E3 4.95	E3 4.95	E3 4.95	E3 4.95	E2 4.40	E1 3.60	PLF

Type U-235	Type U-235 Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	10	4.40
G1	10	4.40 + 8.00
G2	4	4.40 + 9.00
E3	60	4.95

A11TP-4694L-14GV90 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE N

Figure 4.3-9-145, Rev. 0

PLF	E1 3.60	E2 4.00	E3 4.40	E3 4.40	E3 4.40	E3 4.40	E3 4.40	E2 4.00	E1 3.60	PLF
E1 3.60	G3 4.00 10.00	E4 4.95	E4 4.95	E4 4.95	E4 4.95	E4 4.95	G2 4.00 9.00	E4 4.95	G1 4.00 6.00	E1 3.60
E2 4.00	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.00
E3 4.40	E4 4.95	E4 4.95	PLF	E3 4.40	PLF	E3 4.40	PLF	E4 4.95	G2 4.00 9.00	E2 4.00
E3 4.40	E4 4.95	E4 4.95	E3 4.40				G3 4.00 10.00	E4 4.95	E4 4.95	E3 4.40
E3 4.40	E4 4.95	PLF	PLF		Water Channel		PLF	PLF	E4 4.95	E3 4.40
E3 4.40	E4 4.95	E4 4.95	E3 4.40				G3 4.00 10.00	E4 4.95	E4 4.95	E3 4.40
E3 4.40	G2 4.00 9.00	E4 4.95	PLF	G3 4.00 10.00	PLF	G3 4.00 10.00	PLF	E4 4.95	G2 4.00 9.00	E3 4.40
E2 4.00	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E4 4.95	PLF	E4 4.95	E2 4.00
E1 3.60	G1 4.00 6.00	E4 4.95	G2 4.00 9.00	E4 4.95	E4 4.95	E4 4.95	G2 4.00 9.00	E4 4.95	G3 4.00 10.00	E1 3.60
PLF	E1 3.60	E2 4.00	E2 4.00	E3 4.40	E3 4.40	E3 4.40	E3 4.40	E2 4.00	E1 3.60	PLF

Type
U-235

Type
U-235
Gd_2O_3

Pellet Type	Quantity	U-235 + Gd_2O_3 Concentration (wt%)
E1	8	3.60
E2	10	4.00
G1	2	4.00 + 6.00
G2	6	4.00 + 9.00
G3	6	4.00 + 10.00
E3	22	4.40
E4	38	4.95

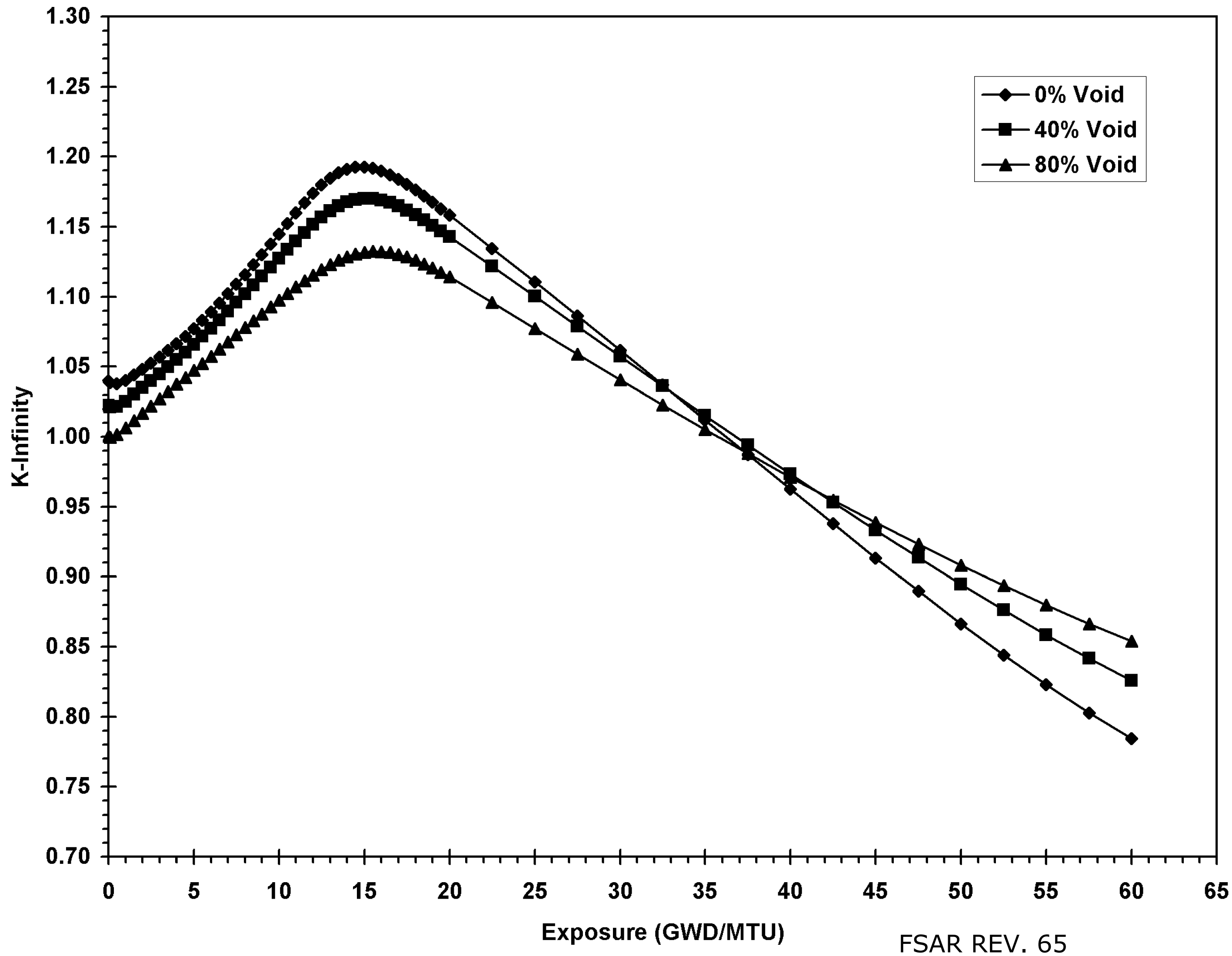
A11T-4462L-14GV100 Enrichment Distribution

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL RADIAL ENRICHMENT
(NOMINAL)
TYPICAL LATTICE O

Figure 4.3-9-146, Rev. 0



FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM^{TM}-10 FUEL DOMINANT LATTICE
HOT - UNCONTROLLED
K-INFINITY VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-11-1, Rev. 55

Auto Cad: Figure Fsar 4_3_11_1.dwg

1.203	1.223	1.209	1.220	1.201	1.198	1.211	1.289	1.041	1.241
1.223	1.070	0.403	0.952	0.885	0.959	0.403	1.080	1.172	1.262
1.209	0.403	0.809	0.844	0.388	0.909	0.884	0.858	0.412	1.232
1.220	0.952	0.844	0.935	1.058	1.170	1.130	0.990	0.953	1.220
1.201	0.885	0.388	1.058	W	W	W	1.081	0.403	1.160
1.198	0.959	0.909	1.170	W	W	W	1.164	0.898	1.196
1.211	0.403	0.884	1.130	W	W	W	1.002	0.407	1.222
1.289	1.080	0.858	0.990	1.081	1.164	1.002	0.465	1.075	1.123
1.041	1.172	0.412	0.953	0.403	0.898	0.407	1.075	1.226	1.292
1.241	1.262	1.232	1.220	1.160	1.196	1.222	1.123	1.292	1.249

Maximum Local Power = 1.292

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM^{TM}-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-2, Rev. 55

Auto Cad: Figure Fsar 4_3_11_2.dwg

0.978	1.018	1.060	1.055	1.036	1.040	1.067	1.071	0.903	0.988
1.018	1.028	0.897	0.960	0.885	0.952	0.868	1.030	1.048	1.034
1.060	0.897	0.927	0.894	0.783	0.918	0.913	0.910	0.939	1.087
1.055	0.960	0.894	0.922	1.007	1.052	1.036	0.982	1.020	1.092
1.036	0.885	0.783	1.007	W	W	W	1.065	0.913	1.084
1.040	0.952	0.918	1.052	W	W	W	1.109	0.979	1.089
1.067	0.868	0.913	1.036	W	W	W	1.071	0.949	1.113
1.071	1.030	0.910	0.982	1.065	1.109	1.071	0.973	1.088	0.998
0.903	1.048	0.939	1.020	0.913	0.979	0.949	1.088	1.074	1.049
0.988	1.034	1.087	1.092	1.084	1.089	1.113	0.998	1.049	0.999

Maximum Local Power = 1.113

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
15000 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-3, Rev. 55

Auto Cad: Figure Fsar 4_3_11_3.dwg

0.936	0.954	1.006	1.023	1.020	1.021	1.028	1.003	0.899	0.937
0.954	0.995	0.951	1.013	0.961	1.010	0.943	1.032	0.993	0.955
1.006	0.951	1.009	0.992	0.912	1.000	0.988	0.975	0.964	1.011
1.023	1.013	0.992	1.009	1.044	1.057	1.052	1.030	1.040	1.036
1.020	0.961	0.912	1.044	W	W	W	1.060	0.963	1.040
1.021	1.010	1.000	1.057	W	W	W	1.073	1.001	1.039
1.028	0.943	0.988	1.052	W	W	W	1.053	0.974	1.041
1.003	1.032	0.975	1.030	1.060	1.073	1.053	0.991	1.051	0.960
0.899	0.993	0.964	1.040	0.963	1.001	0.974	1.051	0.999	0.958
0.937	0.955	1.011	1.036	1.040	1.039	1.041	0.960	0.958	0.941

Maximum Local Power = 1.073

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
40000 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-4, Rev. 55

Auto Cad: Figure Fsar 4_3_11_4.dwg

1.225	1.242	1.225	1.254	1.243	1.235	1.234	1.314	1.058	1.262
1.242	1.052	0.355	0.953	0.893	0.963	0.356	1.068	1.163	1.275
1.225	0.355	0.796	0.843	0.343	0.912	0.887	0.849	0.362	1.237
1.254	0.953	0.843	0.951	1.079	1.197	1.156	0.996	0.933	1.234
1.243	0.893	0.343	1.079	W	W	W	1.088	0.354	1.172
1.235	0.963	0.912	1.197	W	W	W	1.170	0.879	1.210
1.234	0.356	0.887	1.156	W	W	W	0.990	0.355	1.229
1.314	1.068	0.849	0.996	1.088	1.170	0.990	0.409	1.044	1.134
1.058	1.163	0.362	0.933	0.354	0.879	0.355	1.044	1.216	1.304
1.262	1.275	1.237	1.234	1.172	1.210	1.229	1.134	1.304	1.260

Maximum Local Power = 1.314

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 0% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-5, Rev. 55

Auto Cad: Figure Fsar 4_3_11_5.dwg

1.154	1.187	1.178	1.171	1.141	1.144	1.172	1.247	1.006	1.194
1.187	1.085	0.473	0.955	0.878	0.958	0.471	1.095	1.175	1.231
1.178	0.473	0.837	0.857	0.453	0.915	0.892	0.877	0.485	1.217
1.171	0.955	0.857	0.928	1.043	1.152	1.113	0.994	0.980	1.197
1.141	0.878	0.453	1.043	W	W	W	1.083	0.474	1.139
1.144	0.958	0.915	1.152	W	W	W	1.171	0.925	1.171
1.172	0.471	0.892	1.113	W	W	W	1.028	0.481	1.205
1.247	1.095	0.877	0.994	1.083	1.171	1.028	0.545	1.112	1.103
1.006	1.175	0.485	0.980	0.474	0.925	0.481	1.112	1.231	1.264
1.194	1.231	1.217	1.197	1.139	1.171	1.205	1.103	1.264	1.212

Maximum Local Power = 1.264

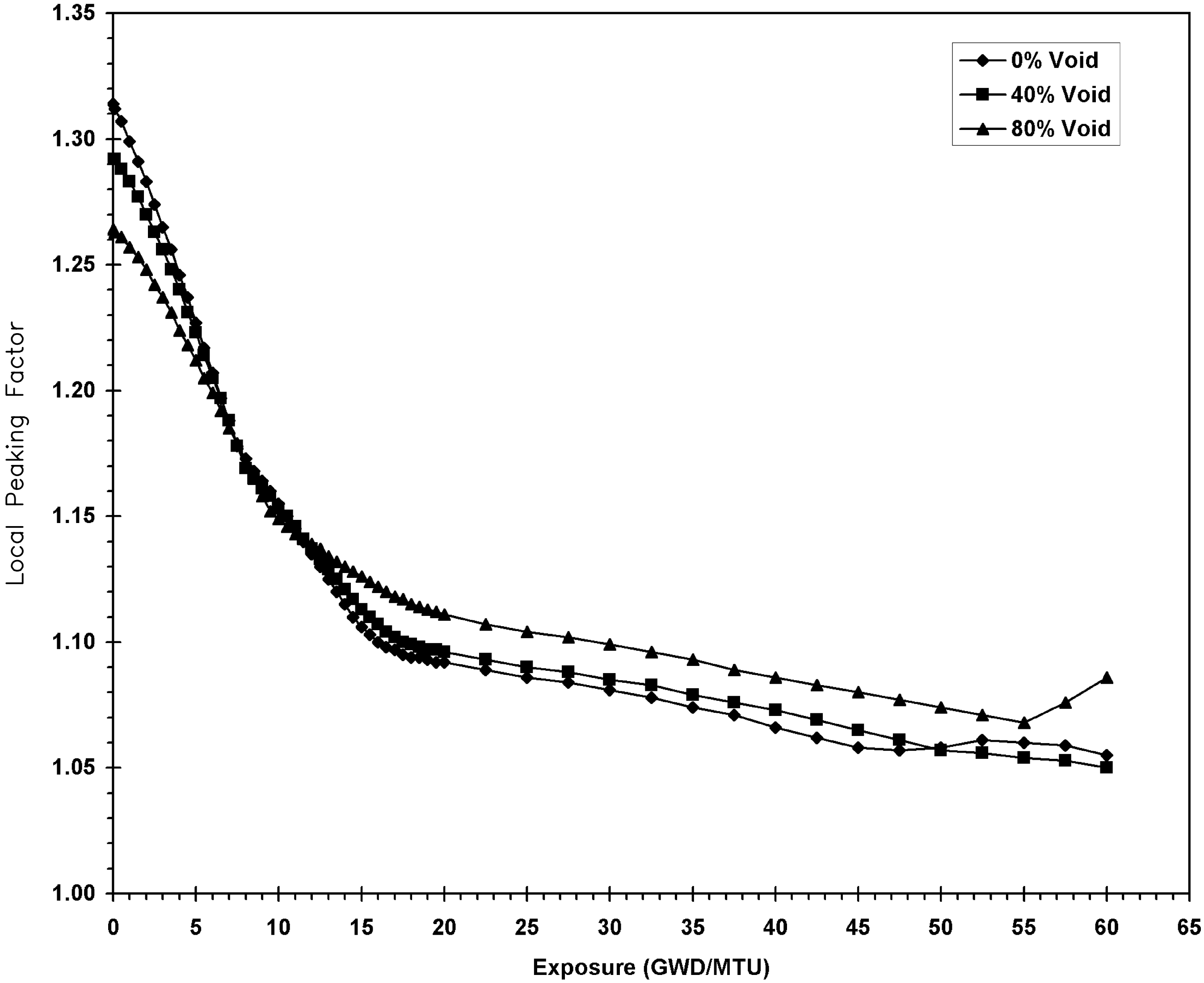
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 80% VOIDS
0 MWD/MTU
(TYPICAL)

FIGURE 4.3-11-6, Rev. 55

Auto Cad: Figure Fsar 4_3_11_6.dwg



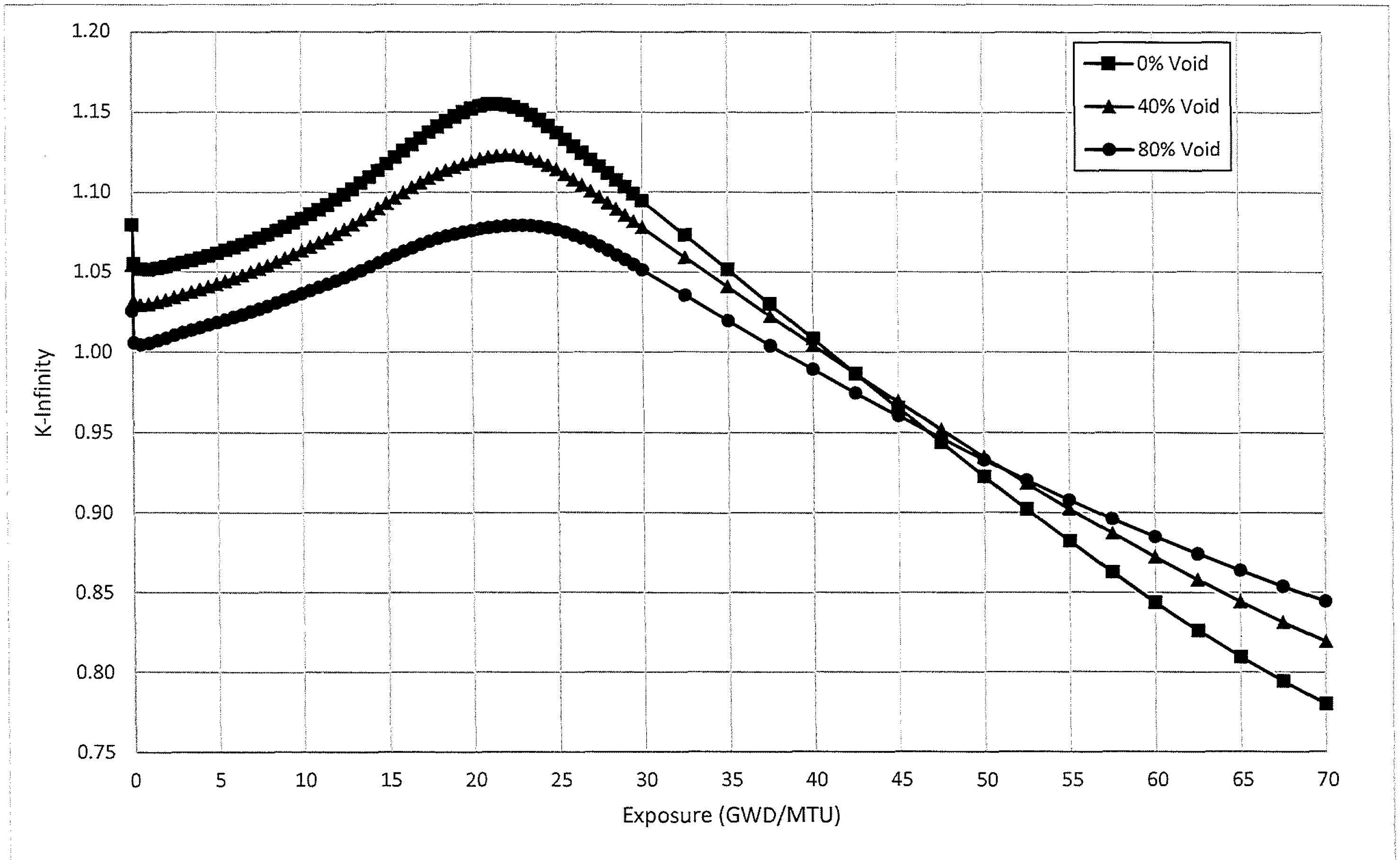
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

FANP ATRIUM™-10 FUEL DOMINANT LATTICE
MAXIMUM HOT-UNCONTROLLED
LOCAL PEAKING FACTOR VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-11-7, Rev. 55

Auto Cad: Figure Fsar 4_3_11_7.dwg



FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL DOMINANT LATTICE HOT – UNCONTROLLED K-INFINITY VS. EXPOSURE (TYPICAL)

FSAR FIGURE 4.3-11-8, Rev. 0

1.110	1.277	1.422	1.421	1.399	1.378	1.322	1.279	1.290	1.181	1.031
1.277	0.399	1.120	1.134	1.138	1.120	1.028	0.367	0.951	0.377	1.172
1.422	1.120	1.035	1.046	1.074	1.068	0.975	0.860	0.851	0.938	1.267
1.421	1.134	1.046	1.083	1.199	1.241	1.092	0.888	0.807	0.359	1.184
1.399	1.138	1.074	1.199	W	W	W	0.364	0.831	0.941	1.238
1.378	1.120	1.068	1.241	W	W	W	1.035	0.896	0.985	1.251
1.322	1.028	0.975	1.092	W	W	W	0.356	0.809	0.926	1.224
1.279	0.367	0.860	0.888	0.364	1.035	0.356	0.769	0.762	0.353	1.208
1.290	0.951	0.851	0.807	0.831	0.896	0.809	0.762	0.800	0.906	1.235
1.181	0.377	0.938	0.359	0.941	0.985	0.926	0.353	0.906	0.367	1.140
1.031	1.172	1.267	1.184	1.238	1.251	1.224	1.208	1.235	1.140	1.001

Maximum Local Power = 1.422

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
0 MWD/MTU
(TYPICAL)

FSAR FIGURE 4.3-11-9, Rev. 0

1.031	1.131	1.220	1.205	1.183	1.176	1.164	1.169	1.183	1.099	1.006
1.131	1.003	1.068	1.018	1.008	1.002	0.975	0.694	1.007	0.907	1.097
1.220	1.068	0.979	0.953	0.965	0.970	0.934	0.893	0.912	1.000	1.176
1.205	1.018	0.953	0.976	1.063	1.107	1.035	0.918	0.873	0.669	1.121
1.183	1.008	0.965	1.063	W	W	W	0.734	0.888	0.946	1.141
1.176	1.002	0.970	1.107	W	W	W	1.054	0.916	0.959	1.142
1.164	0.975	0.934	1.035	W	W	W	0.696	0.873	0.937	1.135
1.169	0.694	0.893	0.918	0.734	1.054	0.696	0.873	0.849	0.650	1.143
1.183	1.007	0.912	0.873	0.888	0.916	0.873	0.849	0.885	0.980	1.162
1.099	0.907	1.000	0.669	0.946	0.959	0.937	0.650	0.980	0.874	1.083
1.006	1.097	1.176	1.121	1.141	1.142	1.135	1.143	1.162	1.083	0.994

Maximum Local Power = 1.220

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
15000 MWD/MTU
(TYPICAL)

FSAR FIGURE 4.3-11-10, Rev. 0

0.937	0.967	1.040	1.052	1.048	1.051	1.057	1.069	1.056	0.978	0.943
0.967	0.941	1.002	0.978	0.974	0.977	0.988	0.921	1.024	0.957	0.978
1.040	1.002	0.967	0.950	0.956	0.965	0.967	0.972	0.988	1.025	1.058
1.052	0.978	0.950	0.965	1.010	1.035	1.022	0.987	0.978	0.926	1.054
1.048	0.974	0.956	1.010	W	W	W	0.958	0.985	0.997	1.066
1.051	0.977	0.965	1.035	W	W	W	1.062	0.983	0.991	1.064
1.057	0.988	0.967	1.022	W	W	W	0.967	0.988	0.998	1.067
1.069	0.921	0.972	0.987	0.958	1.062	0.967	1.003	0.983	0.928	1.077
1.056	1.024	0.988	0.978	0.985	0.983	0.988	0.983	0.993	1.029	1.062
0.978	0.957	1.025	0.926	0.997	0.991	0.998	0.928	1.029	0.962	0.982
0.943	0.978	1.058	1.054	1.066	1.064	1.067	1.077	1.062	0.982	0.945

Maximum Local Power = 1.077

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 40% VOIDS
40000 MWD/MTU
(TYPICAL)

FSAR FIGURE 4.3-11-11, Rev. 0

1.112	1.275	1.436	1.450	1.439	1.421	1.347	1.278	1.281	1.160	1.016
1.275	0.352	1.120	1.155	1.172	1.156	1.042	0.322	0.923	0.327	1.149
1.436	1.120	1.055	1.075	1.114	1.109	1.002	0.862	0.848	0.909	1.253
1.450	1.155	1.075	1.119	1.248	1.286	1.126	0.888	0.806	0.313	1.175
1.439	1.172	1.114	1.248	W	W	W	0.319	0.837	0.940	1.248
1.421	1.156	1.109	1.286	W	W	W	1.042	0.912	1.001	1.271
1.347	1.042	1.002	1.126	W	W	W	0.310	0.811	0.924	1.231
1.278	0.322	0.862	0.888	0.319	1.042	0.310	0.749	0.753	0.306	1.192
1.281	0.923	0.848	0.806	0.837	0.912	0.811	0.753	0.790	0.870	1.211
1.160	0.327	0.909	0.313	0.940	1.001	0.924	0.306	0.870	0.316	1.107
1.016	1.149	1.253	1.175	1.248	1.271	1.231	1.192	1.211	1.107	0.975

Maximum Local Power = 1.450

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 0% VOIDS
0 MWD/MTU
(TYPICAL)

FSAR FIGURE 4.3-11-12, Rev. 0

1.095	1.273	1.394	1.376	1.343	1.319	1.285	1.270	1.287	1.194	1.032
1.273	0.466	1.122	1.114	1.103	1.084	1.019	0.431	0.990	0.447	1.187
1.394	1.122	1.017	1.018	1.034	1.027	0.951	0.869	0.862	0.978	1.268
1.376	1.114	1.018	1.049	1.143	1.192	1.059	0.904	0.815	0.423	1.179
1.343	1.103	1.034	1.143	W	W	W	0.430	0.833	0.947	1.213
1.319	1.084	1.027	1.192	W	W	W	1.033	0.888	0.973	1.212
1.285	1.019	0.951	1.059	W	W	W	0.422	0.814	0.934	1.202
1.270	0.431	0.869	0.904	0.430	1.033	0.422	0.812	0.780	0.418	1.208
1.287	0.990	0.862	0.815	0.833	0.888	0.814	0.780	0.815	0.948	1.242
1.194	0.447	0.978	0.423	0.947	0.973	0.934	0.418	0.948	0.440	1.161
1.032	1.187	1.268	1.179	1.213	1.212	1.202	1.208	1.242	1.161	1.008

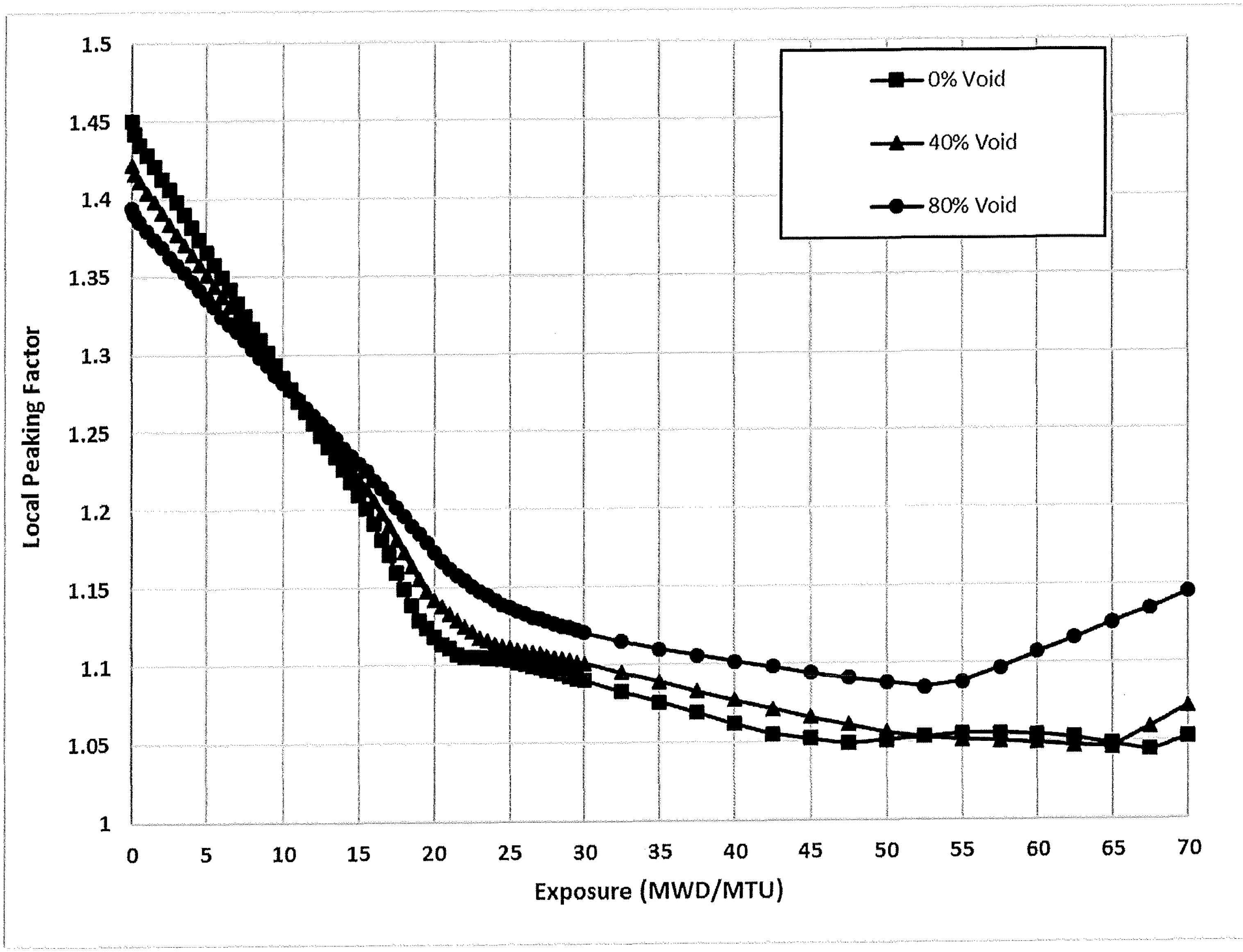
Maximum Local Power = 1.394

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

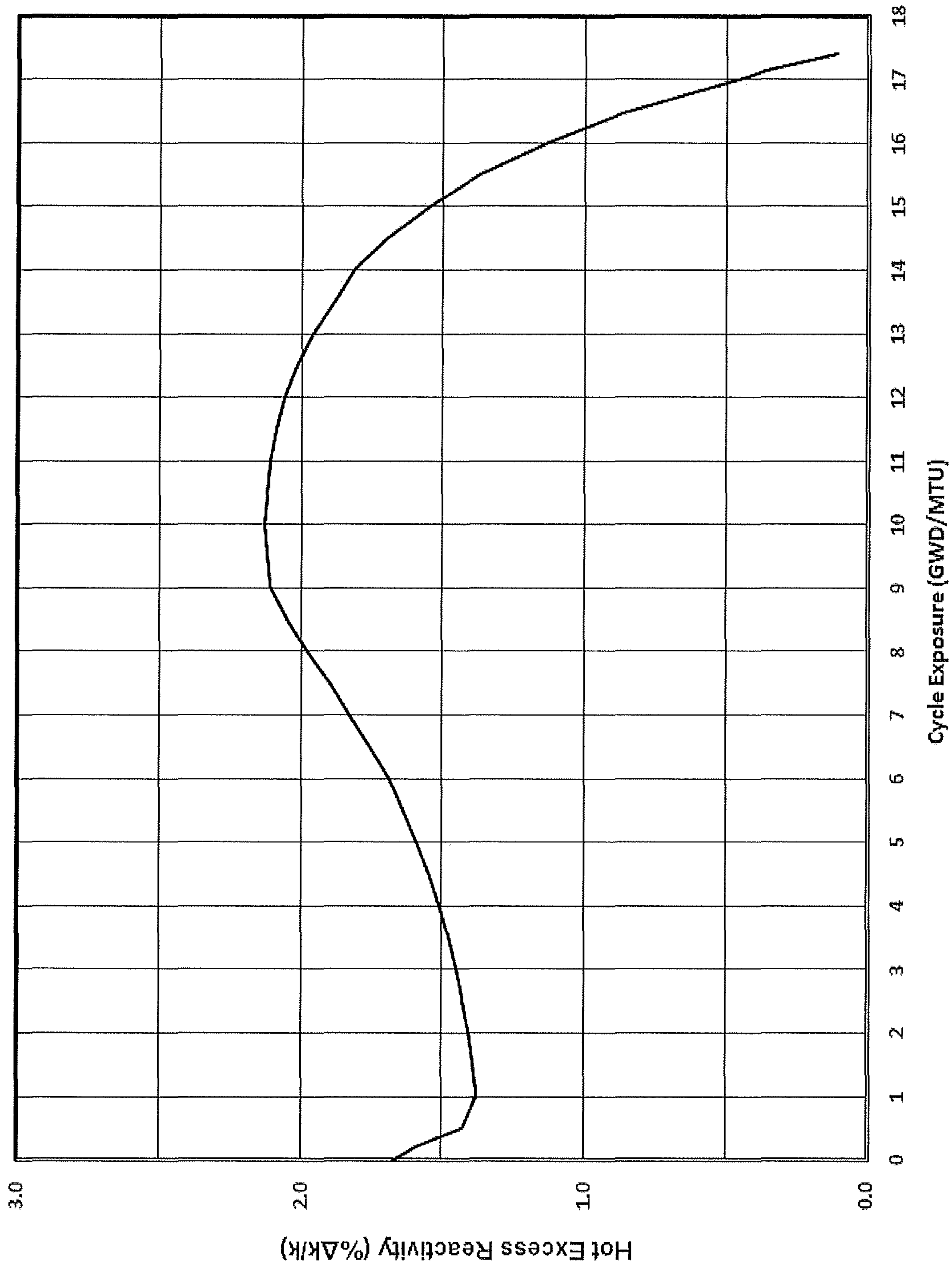
ATRIUM 11 FUEL DOMINANT LATTICE
LOCAL PEAKING, UNRODDED, 80% VOIDS
0 MWD/MTU
(TYPICAL)

FSAR FIGURE 4.3-11-13, Rev. 0



FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
ATRIUM 11 FUEL DOMINANT LATTICE MAXIMUM HOT-UNCONTROLLED LOCAL PEAKING FACTOR VS. EXPOSURE (TYPICAL) FSAR FIGURE 4.3-11-14, Rev. 0



FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

HOT EXCESS REACTIVITY VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-12, Rev. 55

Auto Cad: Figure Fsar 4_3_12.dwg

FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SHUTDOWN MARGIN VS. EXPOSURE
(TYPICAL)

FIGURE 4.3-13, Rev. 55

Auto Cad: Figure Fsar 4_3_13.dwg

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
VESSEL FLUENCE (R.θ) MODEL FOR AZIMUTHAL FLUX DISTRIBUTION FIGURE 4.3-14