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DATA ITEM S-100

**RADIATION AND SHIELDING
ANALYSIS REPORT
NUCLEAR SUBSYSTEM**

VOLUME II CHAPTERS 7 THRU 11

MASTER

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DATA ITEM S-100

RADIATION AND SHIELDING ANALYSIS REPORT, NUCLEAR SUBSYSTEM.

VOLUME II CHAPTERS 7 THRU 11

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7.0 INTERNAL SHIELD

7.1 INTRODUCTION

The Internal Shield consists of a Lead and Boron Carbide-Aluminum-Titanium Hydride (BATH) central shield located forward of the core extending in radius to the outer radius of the core, and peripheral shield located above the reflector extending from the inner radius of the reflector to the inner radius of the pressure vessel. The history of the shield design is reported in the E-105 Data Item.

The sections which follow describe the radiation analysis of the Internal Shield, including comparisons of the calculated environment with the ANSC specified shield criteria. Further, a shield design is recommended to meet these criteria.

7.1.1 Description of the Internal Shield

The central shield is made up of hexagonal cross section BATH modules. BATH is an extruded material consisting of a boron carbide, aluminum and titanium hydride mixture. The across flats dimension of the module is approximately two inches and the length is 11.1 inches. In the peripheral region of the central shield, irregular modules required to form the circumference are cut from the regular hexagonal shaped modules. Coolant channels are provided by grooving 3 faces of each module and drilling 6 axial holes through the module. Assembly of the modules to form the central shield by locating grooved faces against ungrooved faces effectively provides flow channels on all faces. Modules are arranged in arrays of seven. The central module is located by a stud through the cover plate, lead cluster, BATH module and into the support plate.

The six surrounding modules are pinned to the support plate. Close tolerances on hexagonal envelope and pin hole locations minimize gaps between adjacent modules.

The lead cluster, 1.3 inches thick, is shaped to cover a seven BATH module array. The lead cluster is retained by a stud which passes through the cover plate, lead cluster, central BATH module of the array and into the support plate.

Flow channels in the lead align with those in the BATH modules. These flow channels are sized to compensate for the difference in CTE between the lead and BATH material.

The peripheral shield BATH modules are rectangular shapes (sections of rings). Four circumferential rows of modules make up the peripheral shield. Each module is grooved on either 2 or 3 faces, depending upon its location in the shield. Modules on the innermost row of the peripheral shield are also provided with axial coolant channels. Assembly of modules is such that coolant channels are provided at all module faces. Special shaped modules surround the propellant flow and bypass feed pipes.

Eighteen lead sectors, 1.3 inches thick, cover the BATH modules circumferentially between the control drum drive shaft penetrations and radially from the feed and bypass pipes to the support ring. Lead filler blocks cover the outer row of BATH modules between the feed and bypass pipes and piping cylinder.

The central shield is supported by an 0.805-inch thick aluminum plate. The plate is attached at its outer periphery to a radial flange on the piping cylinder (See Chapter 6 Support Plate and Plena for description). Load transmission is then from the shield support plate to the piping cylinder plenum closure ring and thence into the reactor core load path. The peripheral shield is supported by the forward radial flange of the piping cylinder.

7.1.2 Content of Chapter

This chapter provides the radiation analysis data related to the Internal Shield ECC. Section 2.4 previously provided the following for the different types of technical data presented in this chapter:

- 1) A description of the analytical procedures,
- 2) A description of the analytical models, and
- 3) A description of the technique employed to obtain the uncertainty factors for radiation data and the uncertainty factors obtained by this technique.

In addition, the Internal Shield requirements related to radiation analysis are discussed, as well as the status/maturity of all data.

The remaining technical information in this chapter related to the reference design shield can be divided into the following two major areas: 1) radiation data for the Internal Shield at steady state conditions* and 2) radiation data for the Internal Shield at transient conditions. The radiation analysis reported herein was performed for the NSS containing both a composite fueled core and a graphite fueled core.

The radiation data for the Internal Shield in this chapter consists of the following information:

- 1) the maximum radiation environment in the Internal Shield,
- 2) the maximum heating rate in each part of the Internal Shield,
- 3) the heating rate distributions in the heating rate model (See Section 2.4.2) regions which comprise the Internal Shield,
- 4) a summary of the maximum effects of Test Cell C on the Internal Shield environment,
- 5) the photon source distribution in the CRAM model (See Section 2.4.2) regions which comprise the Internal Shield
- 6) the transient radiation environment in the Internal Shield at selected times during the reactor operating profile, and
- 7) a discussion of the probability of failure analysis related to the Internal Shield.

In addition, Section 7.5.3 presents analysis related to the design of an Internal Shield that meets the requirements of the shield for a composite fueled core.

*The steady state conditions referred to are for the NSS with: 1) fuel in the Start of Life (SOL) condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at the End of Life (EOL).

7.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section presents some of the high lights of the Internal Shield analysis for both a composite and graphite fueled Nuclear Subsystem.

7.2.1 External Radiation Environment

The primary function of the Internal Shield is to limit the radiation leakage 1) through a plane 63 inches forward of the core center (perpendicular to the engine axis), and, 2) at critical components in the engine system. Specifically, the "critical components" are three locations pertaining to the lower thrust structure and one location pertaining to the turbopump. The radiation leakage values specified by ANSC are given in Tables 2-1 and 2-2 of this report in terms of photon kerma rate and fast, intermediate and thermal neutron flux.

The radiation environment on the plane and at the critical components was calculated for the R-1 reference design employing the BATH-lead Internal Shield. (The detailed analysis is presented in Section 2.5.1.1.3 of Chapter 2.0.) These results show that all the specifications are met with one exception--the intermediate neutron specification out to a radius of approximately 13 inches on the tangent plane which is approximately 16 percent higher than the specification. However, the calculated gamma radiation environment is about a factor of two lower than the specification values on the plane. This means that more shielding thickness of BATH at the centerline is needed to meet the intermediate neutron specification while lead could be removed and stay within the other requirements.

The above results are basically true for either a composite or graphite fueled Nuclear Subsystem because: 1) no significant difference in neutron fluxes external to the reactor were observed, and, 2) the gamma ray dose rate for the composite fueled core is only about 15 percent less than for the graphite fueled core.

The calculated radiation environment on the plane and at the critical component locations apply to the nominal calculations adjusted both by 1 σ uncertainties and by factors to account for radiation streaming through penetrations in the Internal Shield. The 1 σ uncertainties for photons and neutrons are 1.4 and 3.1, respectively. The factors to account

for radiation streaming depend upon the location of the critical component and/or radius on the plane. Briefly, these factors for photons range from 1.2 to 1.5, and for neutrons from 1.5 to 2.0.

The locations of the Internal Shield specifications are limited to short distances from the reactor. The largest distance from the center of the core is less than 9 feet, i. e. , at the turbopump location. Hence, radiation streaming at these distances is not as significant as that which might occur at large distances from the reactor.

No radiation streaming factor was applied to the calculated environment at the turbopump location. The TPA specification location is at a radius of 36 inches, while the outer radius of the pressure vessel is only approximately 27.8 inches. Hence, the radiation incident at the TPA location is primarily due to radiation leakage through the side surface of the subsystem, rather than through the top surface. Specifically, the two-dimensional discrete ordinates analysis shows that about 80 percent of the total neutron flux and photon dose rate at the TPA is due to side leakage.

It was pointed out above that the internal shield mass thicknesses should be modified to include more BATH and less lead in order to meet the intermediate specification on the plane. (If the specification on the plane is met, calculations have been demonstrated that the specification at the critical components are also met.) The centerline mass thickness of the BATH and lead in the reference design shield is about 104 gm/cm^2 . The radiation analysis computer code model of this shield was comprised of the following approximate weights: BATH, 2160 pounds; lead, 950 pounds; and structure, 260 pounds (total weight of approximately 3370 pounds).

Analysis described in Section 7.5.3 identifies a shield that meets the specifications in all respects. This shield is defined as an "all BATH" shield, having a centerline mass thickness of approximately 87 gm/cm^2 of BATH material (not including structure). Assuming a BATH shield region density of 2.60 gm/cc , which implies a coolant fraction of 0.0887 in the central shield, the centerline thickness is 13.2 inches. Adding the current aluminum forward and aft end plate thickness of 1.2 inches, the total shield assembly thickness would be 14.4

inches and the total weight would be about 2850 pounds. However, this shield thickness of 14.4 inches exceeds the allotted thickness of 13.6 inches.

Additional analysis was performed to define a shield within the 13.6-inch length allocation to meet the specifications. As a result of this study the following recommended shield was defined: 12.15 inches of BATH, 0.25 inches of lead, and 1.2 inches of aluminum structure on the centerline. This shield weighs approximately 2815 pounds, including 2427 pounds of BATH, 184 pounds of lead and 204 pounds of aluminum structure. This shield meets all the specifications except the intermediate specification on the 63-inch plane which was exceeded by only 10 percent on the centerline. Since this is the recommended shield design, it is further recommended that relaxation of the intermediate neutron constraint on the 63-inch plane be considered because the total neutron flux specification is met with this shield.

The question remaining to be answered is: In the case of a graphite fueled Nuclear Subsystem, would the same shield as used in a composite core perform as well against the specification? To answer this question, a detailed analysis was also performed for the graphite fueled case. The results (See Section 7.5.3) show that the 14.4 inch BATH shield meets all the specifications with one exception--the gamma ray dose rate on the plane is exceeded by 9 percent. For the required 13.6-inch length, the recommended lead-BATH shield designed for a composite core meets the gamma ray dose rate specification to within ten percent.

One of the major reasons why the Internal Shield mass can be decreased is that the current reference NSS design now includes 15 grams/cm^2 of material forward of the Internal Shield on the centerline, whereas the previous design (939J685E) had only approximately 4 grams/cm^2 of material forward of the Internal Shield.

Any subsequent major changes in the Nuclear Subsystem design would, of course, require further evaluation of the shields defined above. Furthermore, any weight "quoted" for the Internal Shield is very dependent on the diameter of the subsystem.

7.2.2 Internal Radiation Environment

The maximum radiation environment incident on the Internal Shield ECC occurs at the aft surface of the aft shield plate. The fast, intermediate and thermal neutron flux and the gamma ray dose rate are 1.55×10^{14} , 1.25×10^{15} , $9.3 \times 10^{13} \text{ n/cm}^2\text{-sec}$, and

1.3×10^9 Rads-C/hr, respectively, for a composite fueled core. The corresponding fluences and dose for 60 ten-minute (at rated temperature) operating cycles are 5.6×10^{18} , 4.5×10^{19} , and 3.4×10^{18} n/cm², and 1.3×10^{10} Rads-C, respectively.

For a graphite fueled Nuclear Subsystem, the following variations in these data were observed: the fast, intermediate, and thermal fluxes are lower by 5, 10 and 4 percent, respectively; the gamma ray dose rate is, however, 7 percent higher. (The reasons for these changes are given in detail in Section 2.2.)

The following 1σ uncertainties are applicable to the radiation environment incident at the aft face of the Internal Shield: 1.71 for fast and intermediate neutrons, 1.88 for thermal neutrons, and 1.09 for gamma ray dose rate. These uncertainties were developed from the latest available experimental data and are discussed in detail in Section 2.4.4.2 of Chapter 2.0.

7.2.3 Radiation Heating Rates

The maximum heating rate in the Internal Shield ECC occurs on the reactor centerline at the aft face of the BATH material. This heating rate is 4.6×10^3 BTU/hr-in.³ for a composite fueled core and 4.4×10^3 for a graphite fueled core. The 1σ uncertainty factor associated with this heating rate is 1.75. The major part of this heating rate is due to the (n, α) reaction in the boron in the BATH material. Hence, the heating rate decreases rapidly as a function of axial distance through the BATH central shield; specifically the heating rate decreases by an order of magnitude in the first 2 inches of shield thickness.

The total heat deposition in the Internal Shield ECC is 1.65 Mw for the composite fueled core and 1.61 Mw for the graphite fueled core.

7.2.4 Radiation Sources

The total photon source in the Internal Shield is only 1.8×10^{18} MeV/sec, compared to 6.3×10^{19} MeV/sec for all the NSS regions, excluding the core. (The source in the core region is 9.0×10^{20} MeV/sec.) For the graphite fueled core, the total source in the shield is about 10 percent lower.

About 70 percent of the total photon source in the shield results from the low energy (0.478 MeV) gamma accompanying the boron (n, α) reaction in boron carbide in the

BATH material. Since the major source in the shield consists of low energy photons, the shield source contributes only about 10 percent to the total gamma ray dose rate on the reactor centerline at locations external to the NSS.

The total photon source in the lead portion of the internal shield is only 3.5×10^{15} MeV/sec. Eighty-five percent of this source results from the gamma production due to neutron inelastic scattering.

7.3 REQUIREMENTS

This section identifies the requirements for the Internal Shield related to radiation analysis data.

7.3.1 Specification Identification

The current governing specification for the Internal Shield is Specification No. EC-677562.

7.3.2 Applicable Requirements

The applicable requirements for the Internal Shield which are either evaluated from a radiation standpoint or require radiation information to be evaluated are as follows:

3.1.1.1.3.1.1 Shielding. - The Internal Shield shall meet the requirements specified in 3.1.1.1.4 (Nuclear Radiation Shielding Function).

3.1.1.1.3.1.2 Flow Characteristics. - The Internal Shield shall provide axial flow passage. The maximum allowable radial and azimuthal temperature variation at the central shield inlet shall be as specified in 3.1.1.1.3.1.3 (Flow Separation) of EC-677561. The Internal Shield shall accommodate the requirements of 3.1.1.1.5.1.2 (Startup Operations), 3.1.1.1.5.1.3 (Steady State Operations), 3.1.1.1.5.1.4 (Shutdown Operations), and 3.1.1.1.5.1.5 (Cooldown Operations) of CP-677555. The steady-state and throttle hold thermal-hydraulic characteristics of the Internal Shield pertinent to the Normal Mode shall be in accordance with Tables 1 through 4.

3.1.1.1.4 Nuclear Radiation Shielding Function. - The primary performance characteristic of the Internal Shield supporting the requirements of 3.1.1.1.11 (Nuclear Radiation Shielding Function) of CP-677555 are as follows.

3.1.1.1.4.1 Radiation Leakage. - The Internal Shield shall limit the radiation leakage through a plane 63 inches forward of the core center and perpendicular to the engine vertical axis to the levels specified in Table 10. The Internal Shield shall be designed to minimize radiation streaming through coolant piping or other penetrations in the shield to be consistent with the other requirements specified herein.

3.1.1.1.4.2 Critical Component Radiation Protection. - The Internal Shield shall limit the pressure vessel leakage radiation at critical components in the engine system to the levels specified in Table 11.

3.1.1.1.4.3 Internal Shield Design. - The design of the Internal Shield shall be based on:

- (a) Shield leakage shall be calculated on the basis of nominal formulae
- (b) For preliminary design, leakage values shall be increased by one sigma analytical uncertainty before comparison with the limits specified in Tables 10 and 11.

3.1.2.1 Reliability. - The reliability allocation for the Internal Shield shall be 0.999999.

3.1.2.4.2 Induced Environment.

3.1.2.4.2.1 Nuclear Environment - The Internal Shield shall be designed to withstand the induced nuclear environment specified in Table 12.

3.1.2.4.2.3 Thermal Environment - The Internal Shield shall be designed to withstand the thermal environment attendant to the thermal-hydraulic data specified in Tables 1 through 9, 15 and 16.

3.1.2.4.2.4 Combined Environment. - The Internal Shield shall be designed to withstand the combined environment of 3.1.2.4.1 (Natural Environment) and 3.1.2.4.2 (Induced Environment) when applied sequentially, and in combination to produce augmented environmental stresses.

3.3.1.2 Weight. - The target weight for the Internal Shield is 3275 pounds. "

7.4 METHOD OF ANALYSIS - DESIGN PRACTICES

This section briefly describes the analytical methods and models employed in the radiation analysis of the Internal Shield. Also, included in this section is a discussion of the status/maturity of the data. The validity of the radiation analysis and uncertainty factors applicable to the calculated data are developed.

7.4.1 Analytical Procedures

Section 2.4.1 described 1) the overall framework of nuclear, thermal, and structural analyses, 2) the relationship of radiation and shielding computations to this overall framework, and 3) the details of the procedure which was employed to obtain radiation and shielding results for the NSS and all of the ECCs in the NSS which includes the Internal Shield. In addition, Section 7.5.3 includes a description of the procedures employed to define a shield design to meet the shield criteria.

7.4.2 Analytical Models

Section 2.4.2 provides a detailed description of

- 1) the position at which radiation data at selected locations in the NSS are provided (Figure 2-6 and Table 2-3),
- 2) the computational model employed to perform two-dimensional, discrete ordinates transport calculations with the DOT code (Figures 2-7 through 2-10),
- 3) the radiation heating rate model employed to report heating rates (Figures 2-11 and 2-12), and
- 4) the CRAM model employed to report photon source distributions (Figures 2-13 and 2-14).

Table 2-4 provides a description of the regions contained in each model and indicates which model regions are associated with a particular ECC. Table 2-4, Figure 2-11 for composite fuel and Figure 2-12 for graphite fuel illustrate the following regions which comprise the Internal Shield ECC in the heating rate model:

- 1) Region 35, Aft Central Shield Plate
- 2) Region 36, BATH Central Shield
- 3) Region 38, BATH Peripheral Shield I
- 4) Region 39, BATH Peripheral Shield II
- 5) Region 40, BATH Peripheral Shield III
- 6) Region 41, BATH Peripheral Shield IV
- 7) Region 43, Lead Central Shield
- 8) Region 44, Lead Peripheral Shield I
- 9) Region 45, Lead Peripheral Shield II
- 10) Region 46, Lead Peripheral Shield III
- 11) Region 47, Lead Peripheral Shield IV
- 12) Region 48, Peripheral Shield Plate
- 13) Region 49, Shield Plenum

Table 2-36 and Figure 2-13, for the composite core, and Table 2-55 and Figure 2-14, for the graphite core, illustrate the following regions which comprise the Internal Shield ECC in the CRAM:

- 1) Region 15, Aft Central Shield Plate
- 2) Region 16, BATH Central Shield
- 3) Region 17, BATH Peripheral Shield-1
- 4) Region 18, BATH Peripheral Shield-2
- 5) Region 19, BATH Peripheral Shield-3
- 6) Region 20, BATH Peripheral Shield-4
- 7) Region 21, Lead Central Shield
- 8) Region 22, Lead Peripheral Shield-1
- 9) Region 23, Lead Peripheral Shield-2
- 10) Region 24, Lead Peripheral Shield-3
- 11) Region 25, Lead Peripheral Shield-4
- 12) Region 26, Peripheral Shield Plate
- 13) Region 27, Shield Plenum

The radiation analytical model of the reference design shield employed in the computer codes described herein was comprised of the following approximate weights:

BATH, 2160 pounds; lead, 950 pounds; and structure, 260 pounds.

To insure traceability of all computer programs, tapes, and problems employed in the radiation analysis reported herein, Reference 1 provides the following information:

- 1) Reference Problem Number
- 2) Computer Program Name
- 3) Problem Description
- 4) Input Tape Identification, and,
- 5) Output Tape Identification where applicable.

The major program tapes are also identified in Reference 1.

The Reference Problem Numbers are utilized throughout this chapter to identify computer output data related to graphs and tables wherever possible. These identification numbers are located in the lower right-hand corner of the appropriate graph or table.

7.4.3 Status/Maturity of Input Data

Section 2.4.3 described the type of input information required to perform the radiation analyses of the NSS which, of course, includes the Internal Shield ECC. That section thus described the status/maturity of the input data and the effect on radiation analyses.

The input to radiation analysis for the Internal Shield consisted of atom densities which were based on weights calculated from WANL Drawing 940J419. Subsequent to the time at which these weight calculations were performed, the drawing for the Internal Shield was updated to WANL Drawing 940J440. This layout is similar to 940J419 except as follows: 1) the "Z" ring support was eliminated, 2) the "O" ring seal was eliminated since the "Z" ring and its flange was removed, 3) the central shield aft support plate was reduced in thickness by 0.195 inches, and, 4) the central shield is now supported by a radial flange on the piping cylinder.

In eliminating the internal shield Z-Ring Support, the radius of the central shield was increased by approximately 0.1 inch. Since BATH and the aluminum Z-Ring have similar densities, the net mass change was negligible. The elimination of the Z Ring represents an increase in the central shield material of only 18 pounds. Therefore, this revision to the drawing has a negligible effect on the calculated radiation data for the Internal Shield.

Since the shield support plate thickness was decreased by 0.195 inches, the BATH central shield thickness (presented in Section 7.5.3) must be increased by roughly the same amount in order to meet the gamma ray shield specifications for a composite fueled core.

Section 2.4.2 previously described the shield region models employed in the radiation analysis. Basically, the shield region can be observed in Figures 2-11 and 2-13 as employed in the discrete ordinates transport code and point kernel code, respectively. Both codes employed "smeared" material regions throughout the analysis. (Each region is an annulus rotated about the Z axis of the model.) The actual void percent as a function of radius on a traverse at the aft face of the shield ($Z = 190.58$ cm on Figure 2-13) is given in Figure 7-1, whereas percent of non-BATH material as a function of radius is presented in Figure 7-2.

7.4.4 Validity of Analysis

7.4.4.1 Comparison with Experimental Results

Section 2.4.4.1 described all of the experiments which were performed on the PAX reactor for which calculated and experimental results were compared. Section 2.4.4.1 also provided the major conclusions that were reached for each experiment. Of the experiments performed, the Nuclear Subsystem Shielding Provision Study Experiments⁽²⁾ provided results which applied to the Internal Shield ECC.

In addition to the experiments described in Section 2.4.4.1, one other experiment basically related to the internal shield was performed.⁽³⁾ Briefly, experimental data were obtained on the PAX-G2 reactor using a 60° sector of an R-1 type support plate. The radiation environment that was obtained consists of the neutron absorption rate of ^{164}Dy , the fission

rate of ^{238}U , the reaction rate of $^{32}\text{S} (n,p) ^{32}\text{P}$, the gamma Kerma rate of CaF_2 , and neutron tissue Kerma rate obtained with silicon diode detectors (Phylatrons).

Data were obtained on three experimental configurations:

1. A "dry" 60° sector. This nomenclature denotes the configuration with the R-1 60° support plate sector without polyethylene and stainless steel mockup of the liquid hydrogen (LH_2) and cluster support hardware. The remaining 300° sector was a slab mockup of the 60° sector.

2. A "wet" 60° sector. This nomenclature denotes the configuration with the R-1 60° support plate sector that includes the polyethylene and stainless steel mockup of the LH_2 and cluster support hardware. The remaining 300° sector was a slab mockup of the 60° sector.

3. A "wet" 360° sector. This nomenclature denotes the configuration that is a 360° mockup of the wet 60° sector preserving the same areal density (gm/cm^2) for Al, $(\text{CH}_2)_n$, and stainless steel as in the 60° sector.

These three configurations also had a BATH internal shield mockup placed on top of the support plate configurations. The total thickness of the BATH mockup approximated an 8-inch BATH shield.

The purpose of these experiments was to establish the effect of a "smeared" model (simulated by slabs of material) versus an explicit model of the support plate on the radiation environment transmitted to the BATH internal shield. This effect could not readily be determined analytically with currently available computer techniques; thus, experiments were performed to derive empirical data. To obtain empirical results, all of the data were analyzed on a relative basis using the "wet" 360° configuration as the base case. The relative experimental uncertainties were used to statistically analyze the results.

The experimental data were not compared with neutron or photon transport calculations in this evaluation. The comparison, however, can be used to determine empirical "correction" factors to be applied to transport calculations of the radiation environment which use a "smeared" geometry model of the support plate.

From the initial analysis of the experimental data, the following conclusions are made:

1) A comparison of the 360° slab data with the 60° support plate data show that within the statistical uncertainty of the data no radiation streaming through the individual penetrations in the 60° sector was observed.

2) The use of a "smeared" slab mockup of the "wet" support plate underestimates the boron (n, α) heating rate by a factor of 1.08 to 1.263, 0.85 to 1.41, and 0.88 to 1.455 in the dosimeter slots 3, 4, and 5, respectively, in the BATH shield mockup. These are based upon the results that 90 percent of the data will yield this range of values with 95 percent probability. These conclusions are based upon ^{164}Dy activation.

3) The ^{164}Dy activation was the only dosimeter that yielded a statistically significant difference in the radiation environment at the aft face of the central shield when the 360° slab mockup and 60° support plate measurements were compared.

4) For the "dry" support plate, the results for all of the dosimeters show that there is no significant difference in the radiation environment at the aft face of the central shield mockup for the 60° sector or a slab mockup of the sector. This was the observation for all dosimeter types used within the quoted relative uncertainty of the results.

5) The ratio of the radiation environment at the aft face of the central BATH shield mockup (dosimeter slot 3 within the radius of the support plate) without $(\text{CH}_2)_n$ and stainless steel to that with the support plate with these materials in a slab array is $4.79 \times 10^{-2} \pm 0.39 \times 10^{-2}$ for ^{164}Dy activation (proportional to boron (n, α) heating, 1.0 (within statistical uncertainty), for CaF_2 (photon Kerma rate), 2.10 ± 0.16 for $^{32}\text{S}(\text{n},\text{p})^{32}\text{P}$ reaction rate (fast neutron flux for $E > 2.9$ MeV), 1.86 ± 0.22 for ^{238}U fission rates (fast neutron flux for $E > 1.5$ MeV), and 2.21 ± 0.40 for the Phylatron dosimeters (fast neutron Kerma rate of tissue). These ratios are equal within the statistical uncertainty of the data for dosimetry slots within the shield mockup except for the ^{164}Dy activation and CaF_2 photon Kerma rate. The ratio of "dry" to "wet" for ^{164}Dy within the shield mockup is 2.13 ± 0.11 . The difference between the slot 3 and 4 ratio is due to the presence of boron between slots 3 and 4. The

ratio in slots 6 and 7 for the CaF_2 dosimeters is 1.4 ± 0.18 . This change in the ratio from the previous slots is possibly due to a combination of: 1) response of TLDs to low energy neutrons, 2) local secondary gamma ray production, and 3) photon spectral difference of "wet" and "dry" environment.

It is observed from the results of the experiment that small differences in the radiation environment were measured between a slab mockup and a 60° support plate sector. To reduce the uncertainties associated with the results, it was recommended⁽³⁾ that:

- 1) The experiments be repeated several times to reduce random uncertainties.
- 2) A full 360° support plate sector should be used.

7.4.4.2 Uncertainty Analysis

The results for the analysis of statistical variations between calculated and experimental data for the Nuclear Subsystem Shielding Provision Study Experiments are presented in Section 2.4.4.2. The derivation of uncertainty factors for the internal radiation environment data (fast, intermediate, and thermal neutron fluxes and gamma ray dose rates) and radiation heating rates (both neutron and photon) via the procedures outlined in NRP 605 are also described in Section 2.4.4.2.

Uncertainty factors are derived for possible variations in the internal radiation data at a point (pointwise uncertainty factor, F_p) and for the possible variations in average radiation data over a region (i.e., a regionwise uncertainty factor, F_R , for the total neutron heating rate in the shield). Tables 2-22 and 2-29 present these uncertainty factors for the Internal Shield regions for composite and graphite fueled cores, respectively. The pointwise factors for the composite core case are 1.71, for fast and intermediate neutron fluxes, 1.88 for thermal neutron fluxes, 1.31 for neutron kinetic heating rates, and 1.09 for photon dose rates and photon heating rates. The regionwise factors for neutron kinetic and photon heating are 1.13 and 1.04, respectively. Similar factors for a graphite fueled core, presented in Table 2-29, are approximately 20 percent less than those for the composite core case as discussed previously in Section 2.4.4.2.

Table 2-11 presents the uncertainty factors employed for the external environment data. The derivation of these factors is described in Section 2.4.4.2.

As described in Section 7.4.4.1, the R-1 support plate experimental data also indicate that the heating rates at the front face of the BATH shield material are probably low by roughly a factor of 1.2.

7.5 RESULTS, ANALYSIS, AND INTERPRETATION

This section presents the detailed results of the radiation analysis of the reference design BATH-lead shield and an interpretation of the significance of these results. An internal shield design to meet the shield radiation requirements is also presented.

7.5.1 Composite Core

The radiation analysis of the Internal Shield was performed for the NSS containing both a composite fueled core and a graphite fueled core. This section presents the results obtained for a composite fueled core.

7.5.1.1 Steady State

This section presents the radiation data for the Internal Shield at steady state conditions defined in Section 2.5.1.1.

7.5.1.1.1 Internal Radiation Environment

Table 7-1 presents the maximum calculated radiation environment in the Internal Shield ECC for the NSS containing a composite fueled core. The maximum environment occurs on the aft face of the aft shield plate on the reactor centerline. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. (It should be noted that the heating rate in Table 7-1 is the maximum gamma heating only and does not include the neutron heating in the BATH material. This latter information is presented in Table 7-2.) Also presented in this table are the neutron fluences and gamma ray doses for one hour operation at full power (1511 Mw) and for 60 ten-minute cycles as defined by the power profile in Section 2.5.1.2. Section 7.5.1.2.1 presents the maximum environment integrated over the nominal reactor operating profile (ten minutes at rated temperature) at several state points during operation.

The radiation environment (fast, intermediate, and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the Internal Shield are presented in Section 2.5.1.1.1. Section 2.5.1.2.1 presents radiation data at selected locations in the Internal Shield integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides the safe radiation levels for the NSS materials which can occur before significant radiation damage occurs. By comparing those levels with the preceding results and the radiation data to be presented in Section 7.5.2.1.1 for a graphite fueled core, the effect of radiation damage in Internal Shield materials can be assessed.

7.5.1.1.2 Radiation Heating Rates

Total Heating Rates

Table 2-49 provides the heating rates for the heating rate model regions (See Section 7.4.2) associated with the Internal Shield ECC for the NSS containing a composite fueled core. This table indicates that 1.65 Mw of heat are deposited in the regions comprising the Internal Shield. Section 2.5.1.1.4 describes data on this table in detail.

Table 7-2 provides the maximum heating rate in each component in the Internal Shield. This table provides the following information:

- 1) The drawing numbers for the Internal Shield ECC,
- 2) The item number of each component in the ECC as given by the drawings,
- 3) The part nomenclature,
- 4) The material of which the part is made, and
- 5) The maximum heating rate for the component (Btu/hr-in^3) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate occurs in the BATH Central Module (Item No. 14), and is $4.60 \times 10^3 \text{ Btu/hr-in}^3$ for the NSS containing a composite fueled core. The heating rate in the same component for the NSS containing a graphite fueled core is about 5 percent lower.

Heating Rate Distributions

Figures 7-3 through 7-44 provide the relative radial and relative axial distribution of radiation heating for each heating rate model region (See Section 7.4.2) which comprises the Internal Shield ECC. These data apply to the NSS containing a composite fueled core.

The heating rate at any location within the shield regions is determined in the following manner:

- 1) Determine the value of the relative radial distribution of heating for the position of interest,
- 2) Determine the value of the relative axial distribution of heating for the position of interest,
- 3) Determine the average value of the heating rate from the appropriate curve (the average value of the heating rate appears on both the radial and axial distribution curves for each region), and
- 4) The heating rate at the position of interest is the product of the average heating rate, the value of the relative radial distribution, and the value of the relative axial distribution.

7.5.1.1.3 Radiation Sources

Table 2-36 provides gamma ray source strengths for each of the regions in the CRAM model which comprise the Internal Shield ECC (See Section 7.4.2). Section 2.5.1.1.2 provides a description of the data presented in this table.

The total photon source in the shield regions is 1.8×10^{18} MeV/sec for the NSS employing a composite fueled core. The total source for the graphite fueled core case is approximately 10 percent lower.

The total source in the shield, however, contributes less than 10 percent to the total dose on the centerline 10 feet from the reactor core, and hence, is not a significant source of radiation.

The data in Figures 7-~~45~~ through 7-70 represent the distributions in the Shield regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occurred with the change in cores.

The photon source strength at any location and for any given energy group in the Internal Shield ECC is determined in the following manner:

- 1) Determine the value of the relative radial distribution of photon source at the position of interest,
- 2) Determine the value of the relative axial distribution of photon source at the position of interest,
- 3) Determine the source strength for the energy group and region of interest from Table 2-36, and
- 4) The source strength for the energy group and position of interest is the product of the source strength, the value of the relative radial distribution, and the value of the relative axial distribution.

7.5.1.1.4 Effect of Test Cell C on Radiation Data

Section 2.5.1.1.5 presents the results, and a description of these results, which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data (Figure 2-153) indicate that the maximum increase in the radiation environment in the shield occurs in the thermal flux near the forward face and is about 25 percent.

7.5.1.1.5 External Environment

Comparisons of calculations with the current shield criteria were previously presented in graphical form on the tangent plane (Figures 2-123 through 2-126) and in tabular form at all the engine specification locations (Tables 2-~~40~~ through 2-44). It was noted that the

intermediate neutron flux criteria on the tangent plane was exceeded by 16 percent out to a radius of 33 centimeters which indicates a need for additional neutron shield material. However, Figure 2-123 and Table 2-40 clearly show that the gamma ray specification is "overshielded" by about a factor of two. This clearly indicates that mass thickness, and hence, weight should be removed from the internal shield to meet the gamma specifications. In other words, the relative neutron to gamma shield material thickness must be modified to meet the specification with a minimum weight shield. The analysis leading to a recommended shield design is presented in Section 7.5.3.

Other external radiation data for the composite fueled core with the lead-BATH reference design shield are given in meridian ring format in Figures 2-127 through 2-130, on the centerline in Figures 2-131 and 2-132, and in isodose and isoflux form in Figures 2-133 through 2-136.

7.5.1.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section 2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

7.5.1.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 7-3 presents the maximum radiation data in the Internal Shield ECC for the NSS with a composite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 7-3 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes these radiation data in more detail, and Table 2-51 provides similar information at selected locations in the Internal Shield ECC.

7.5.1.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-49, to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-49.

7.5.1.3 Probability of Failure

Paragraph 3.3.1.3, Nuclear Criteria, of the NSS Specification CP-677555A requires that 1) the effect of radiation leakage on the calculated radiation levels at the specification locations be included, and 2) the radiation levels which include the effect of streaming are to be increased by the one sigma analytical uncertainty before comparison with the specified radiation levels. The derivation of the one sigma analytical uncertainty according to the procedures outlined in NRP 600 and NRP 605 is described in detail in Section 2.4.4.2 and the results are summarized in Section 7.4.4.2.

The preceding criteria was employed to evaluate the internal shield design. A failure occurs only when the specification radiation levels are exceeded. That is, since the radiation levels for which the internal shield was designed, include the possible one sigma analytical uncertainty, a failure occurs only when the radiation levels at the specification locations are greater than the one sigma upper limit. A failure does not occur when the radiation levels at the specification locations are less than the one sigma lower limit. By the definition of a normal probability distribution, 68 percent of the possible values of the radiation data lies between the one sigma upper limit and the one sigma lower limit; 32 percent of the possible values of the radiation data exists beyond these limits. However, only half, or 16 percent, of the possible values of the radiation data are greater than the one sigma upper limit. Therefore, the probability of failure, or the probability that the radiation levels at the specification locations are exceeded, is 0.16.

7.5.2 Graphite Core

The radiation analysis of the Internal Shield was performed for the NSS containing both a composite fueled core and a graphite fueled core. This section presents the results obtained for a graphite fueled core.

7.5.2.1 Steady State

This section presents the radiation data for the Internal Shield at steady state conditions defined in Section 2.5.2.1.

7.5.2.1.1 Internal Radiation Environment

Table 7-1 presents the maximum radiation environment in the Internal Shield ECC for the NSS containing a graphite fueled core. The maximum environment occurs on the aft face of the aft shield plate on the reactor centerline. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. (It should be noted that the heating rate in Table 7-1 is the maximum gamma heating only and does not include the neutron heating in the BATH material. This latter information is presented in Table 7-2). Also presented in this table are the neutron fluences and gamma ray

doses for one hour operation at full power (1511 Mw) and for 60 ten-minute cycles as defined by the power profile in Section 2.5.1.2. Section 7.6.1.2.1 presents the maximum environment integrated over the nominal reactor operating profile at several state points during operation.

The radiation environment (fast, intermediate, and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the Internal Shield are presented in Section 2.5.2.1.1. Section 2.5.2.2.1 presents radiation data at selected locations in the Internal Shield integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides the safe radiation levels for the NSS materials which can occur before significant radiation damage occurs. By comparing these levels with the preceding results, the effect of radiation damage on Internal Shield materials can be assessed.

7.5.2.1.2 Radiation Heating Rates

Total Heating Rates

Table 2-65 provides the heating rates for the heating rate model regions (See Section 7.4.2) associated with the Internal Shield ECC for the NSS containing a graphite fueled core. This table indicates that 1.61 Mw of heat are deposited in the regions comprising the Internal Shield. Section 2.5.1.1.4 describes data on this table in detail.

Table 7-2 provides the maximum heating rate in each component in the Internal Shield. This table provides the following information:

- 1) The drawing numbers for the Internal Shield ECC,
- 2) The item number of each component in the ECC as given by the drawings,
- 3) The part nomenclature,
- 4) The material of which the part is made, and

- 5) The maximum heating rate for the component (Btu/hr-in³) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate occurs in the BATH Central Module (Item No. 14), and is 4.40×10^3 Btu/hr-in³ for the NSS containing a graphite fueled core. The heating rate in the same component for the NSS containing a composite fueled core is about 5 percent higher.

Heating Rate Distributions

Figures 7-71 through 7-112 provide the relative radial and relative axial distribution of radiation heating for each heating rate model region (See Section 7.4.2) which comprises the Internal Shield ECC. These data apply to the NSS containing a graphite fueled core.

The heating rate at any location within the shield regions is determined in the same manner as described in Section 7.5.1.1.2.

7.5.2.1.3 Radiation Sources

Table 2-55 provides gamma ray source strengths for each of the regions in the CRAM model which comprise the Internal Shield ECC (See Section 7.4.2). Section 2.5.1.1.2 provides a description of the data presented in this table.

The total photon source in the shield regions is 1.6×10^{18} MeV/sec for the NSS employing a graphite fueled core. The total source for the composite fueled core case is approximately 10 percent higher.

The total source in the shield, however, contributes less than 7 percent to the total dose on the centerline 10 feet from the reactor core, and hence, is not a significant source of radiation.

The data in Figures 7-45 through 7-70 represent the distributions in the Shield regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occurred with the change in cores.

The photon source strength at any location and for any given energy group in the Internal Shield ECC is determined in the manner previously described in Section 7.5.1.1.3.

7.5.2.1.4 Effect of Test Cell C on Radiation Data

Section 2.5.2.1.5 presents the results, and a description of these results which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data Figure 2-289 indicate that the maximum increase in the radiation environment in the shield occurs in the thermal flux near the forward face and is only about 25 percent.

7.5.2.1.5 External Environment

Comparisons of calculations with the current shield criteria were previously presented in graphical form on the tangent plane (Figures 2-169 through 2-272) and in tabular form at all the engine specification locations (Tables 2-56 through 2-60). It was noted that the intermediate neutron flux criteria on the tangent plane was exceeded by 16 percent out to a radius of 33 centimeters which indicates a need for additional neutron shield material. However, Figure 2-269 and Table 2-56 show that the gamma ray specification is "overshielded" by about a factor of two. This clearly indicates that mass thickness, and hence, weight should be removed from the internal shield to meet the gamma specifications. In other words, the relative neutron to gamma shield material thickness must be modified to meet the specification with a minimum weight shield. The analysis leading to a recommended shield design is presented in Section 7.5.3.

Other internal radiation data for the graphite fueled core with the lead-BATH reference design shield are given in meridian ring format in Figures 2-127 through 2-130, on the centerline in Figures 2-131 and 2-132, and in isodose and isoflux form in Figures 2-134 through 2-136, and Figure 2-273.

7.5.2.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section

2.5.1.2 describes this profile and how the transient data provided in this section relate to the State Points indicated on the profile.

7.5.2.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 7-4 presents the maximum radiation data in the Internal Shield ECC for the NSS with a graphite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 7-4 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes these radiation data in more detail, and Table 2-66 provides similar information at selected locations in the Internal Shield ECC.

7.5.2.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-65 to the total heat

deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-63.

7.5.2.3 Probability of Failure

For the NSS containing a graphite fueled core, the probability that the internal shield will fail to meet the external radiation specification levels cannot be determined because the internal shield has been designed to meet these radiation levels only for the NSS containing a composite fueled core. If the internal shield were designed to meet the external radiation specification levels for the NSS containing a graphite fueled core, then the probability of failure to meet these radiation levels would be the same as that quoted in Section 7.5.1.3 for the internal shield with NSS containing a composite fueled core.

7.5.3 Shield Design to Meet Current Specifications

The information presented in Chapter 2.0 and in Sections 7.5.1 and 7.5.2 was related to analysis of the reference design shield in the NSS employing both composite and graphite fuel in the core. The reference shield design centerline thickness consisted of 11.1 inches of BATH material, 1.3 inches of lead, and 1.2 inches of aluminum end plates. It was evident from the preceding analysis that this reference design shield over-attenuated the photon radiation on the plane (63 inches forward of the core center) by roughly a factor of two and under-attenuated the intermediate neutron levels by about 16 percent. This section presents the analysis to determine a shield which meets the specification as nearly as possible for the composite fueled core. Data are included which define how much the specifications are exceeded for the graphite fueled case.

7.5.3.1 Method of Analysis

In order to determine the shield mass thickness required to meet the specifications, it is necessary to perform a parametric analysis, i.e., parametric in terms of shield thickness. The parametric analysis was performed using the point kernel program, KAP-V. Results of

the point kernel analysis are shown in Figures 7-113 and 7-114, for gamma rays and fast neutrons, respectively. Plotted in these figures is the following information as a function of BATH shield region thickness:

- 1) the specification on the tangent plane *
- 2) the nominal point kernel calculation
- 3) the nominal calculation multiplied by the 1σ uncertainty (from Table 2-11)
- 4) the nominal calculation multiplied by both the 1σ uncertainty and the factor to account for radiation streaming (defined in Section 2.4.1.3).

It should be noted that these two figures apply only to the tangent plane specification. Based on data previously presented in Section 2.5, it has been determined that if the tangent plane specifications are met, the specifications at all other engine locations are also met. (This will again be demonstrated in the sections which follow). It is further assumed that, in the case of an all BATH shield, if the gamma ray specification is met, then the neutron specifications are also met.

From Figure 7-113, it is evident that a BATH shield thickness of 33.5 cm meets the gamma ray specification on the tangent plane.

To insure that this shield meets the specifications at all engine locations, point kernel neutron and photon problems and DOT-MAP neutron problems were run for the composite core case. The results of these calculations are presented in Tables 7-5 through 7-9 for the centerline of the tangent plane, the TPA, and LTS-1, LTS-2, and LTS-3, respectively. These tables are self-explanatory with the exception of the KAP-V nominal intermediate and thermal flux data. To obtain these data, the DOT-MAP calculated ratios of thermal/fast neutron flux and intermediate/fast neutron flux were multiplied times the KAP-V neutron flux values.

* In this parametric study the center of the core with respect to the vessel dome was changed as the shield thickness varied, i.e., was not fixed at 63 inches.

The data in Tables 7-5 through 7-9 indicate that this shield meets all the specifications.

The question remaining to be answered is: In the case of a graphite fueled Nuclear Subsystem, does this same shield meet all the specifications? To answer this question, the KAP-V and DOT-MAP problems were repeated for the graphite core case. The results are tabulated in Tables 7-10 through 7-14. From these tables, it can be noted that all the specifications are met with one exception, i.e., the gamma ray dose rate on the tangent plane, which is exceeded by about 9 percent.

To meet the specifications, the required central BATH shield thickness is 13.2 inches; the aft aluminum shield support plate employed in this analysis is 1.0 inches and the forward shield aluminum plate thickness is 0.2 inches. The total central shield assembly thickness is 14.4 inches including the end plates. The following coolant fractions were employed in this shield design: central BATH shield, 0.0887; aft end plate, 0.157; forward end plate, 0.198.

A preliminary weight calculation indicates that this all BATH shield weighs approximately 2844 pounds, including 200 pounds for the aluminum end plates.

The peripheral shield would be similar to that shown schematically in Figure 2-6, but with the peripheral lead shielding removed and the total BATH thickness increased by 0.8 inches to correspond to the increase in the central shield assembly thickness.

Since the all BATH shield thickness of 14.4 inches exceeded the allotted internal shield thickness of 13.6 inches, a parametric study of a BATH-lead shield was performed to determine if a 13.6-inch thick BATH-lead shield could meet the specifications. Results of this study are given in Tables 7-15 through 7-19. The BATH-lead shield has the following centerline thickness: 12.15 inches of BATH, 0.25 inches of lead, and 1.2 inches of aluminum structure (total thickness = 13.6 inches). From Tables 7-15 through 7-19, it is observed that all the specifications are met with the exception of the intermediate neutron specification which is exceeded by only 10 percent on the centerline of the 63-inch plane. This shield weighs approximately 2815 pounds which

is about 500 pounds less than the reference design shield.

The 13.6-inch thick BATH-lead shield is the recommended shield design. Since the total neutron specification on the 63-inch plane is met, it is further recommended that relaxation of the intermediate neutron specification on the 63-inch plane be considered.

7.6 REFERENCES

1. M. A. Capo, "Computer Program Traceability for the R-1 Reference Design (939J723) Analysis," DRM 52850, (December, 1970).
2. S. L. Anderson and J. R. Magaw, "Nuclear Subsystem Shielding Provision Study Experiments," WANL-TME-1912, (September, 1970).
3. V. S. Oblock and J. M. Cardito, "Initial Analysis of Radiation Measurements Performed on a 60 Degree R-1 Type Support Pate," DRM 52844, (September, 1970).

TABLE 7-1
MAXIMUM RADIATION ENVIRONMENT AT INTERNAL SHIELD
COMPOSITE AND GRAPHITE FUEL IN THE REACTOR CORE

Description	Location		Gamma Environment			
	Radius (in.)	Reactor Station (in.)	Heating Rate (w/gm-SS)	Dose Rate (Rads-C/hr)	1 Hour Dose (Rads-C)	10 Hour Dose (Rads-C)
Composite Core						
Aft Surface of Aft Shield Plate - Reactor Centerline	0.000	-10.830	4.16(0)	1.34(9)	1.34(9)	1.34(10)
Graphite Core						
Aft Surface of Aft Shield Plate - Reactor Centerline	0.000	-10.830	4.38(0)	1.43(9)	1.43(9)	1.43(10)

Description	Neutron Environment								
	Fast ($E > 1.0$ MeV)			Intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)			Thermal ($E < 0.4 \text{ eV}$)		
	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)
Composite Core									
Aft Surface of Aft Shield Plate - Reactor Centerline	1.55(14)	5.59(17)	5.59(18)	1.25(15)	4.51(18)	4.51(19)	9.32(13)	3.35(17)	3.35(18)
Graphite Core									
Aft Surface of Aft Shield Plate - Reactor Centerline	1.47(14)	5.30(17)	5.30(18)	1.12(15)	4.03(18)	4.03(19)	8.98(13)	3.24(17)	3.24(18)

TABLE 7-2

MAXIMUM HEATING RATE BY COMPONENT FOR THE INTERNAL SHIELD
(940J419)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
1	Stud. 312-24 Unf-2A x 14.44 LG	Al	2.65(2)	2.78(2)
2	Stud. 190-32 Unf-2A x 13.66LG	Al	6.08(2)	6.40(2)
3	Bolt. 190-32 Unf-2A x 12.09LG	Al	1.57(2)	1.68(2)
4	Heli-Coil Insert .312-24	CRES	9.22(2)	9.52(2)
5	Heli-Coil Insert .190-32	CRES	1.87(3)	1.93(3)
6	Nut-.190-32 Unf-2B	CRES	1.22(1)	1.23(1)
7	Nut-.312-24 Unf-2B	CRES	8.28(0)	8.65(0)
8	Locking Device .190	Al	4.09(0)	4.28(0)
9	Locking Device .312	Al	2.78(0)	2.90(0)
10	Locating Pin .250 DIA	Al	6.08(2)	6.40(2)
11	Z Ring, Support	Al	2.65(2)	2.78(2)
12	O-Ring .125 DIA Tubing-.010 Wall	CRES	8.05(0)	8.41(0)
13	Plate, Support 1.000 Thk	Al	6.08(2)	6.40(2)
14	Module, Regular 11.10 LG (Central)	BATH	4.60(3)	4.40(3)
15	Module, Irregular 11.10 LG (Central)	BATH	2.76(3)	2.67(3)
16	Cluster, Regular 1.300 Thk	Lead	1.86(2)	1.93(2)
17	Cluster, Irregular 1.300 Thk	Lead	1.23(2)	1.27(2)
18	Plate, Cover .200 Thk	Al	4.12(0)	4.30(0)
19	Container	Al	1.57(2)	1.68(2)
20	Plate, Support .200 Thk	Al	1.97(2)	2.06(2)
21	Wedges 11.100 Thk (Peripheral)	BATH	4.44(3)	4.25(3)

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 7-2 (Continued)

MAXIMUM HEATING RATE BY COMPONENT FOR THE INTERNAL SHIELD
(940J419)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
22	Wedge Cover 1.300 Thk	Lead	1.25(2)	1.31(2)
23	Ring, Cover	Al	4.43(0)	4.55(0)
24	Sleeve, 1.750 OD	Al	1.57(2)	1.68(2)
25	Ring, Impedance	CRES	1.19(1)	1.22(1)
26	Sector, Cover 1.300 Thk	Lead	1.51(2)	1.55(2)

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 7 - 3

MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE INTERNAL SHIELD
(COMPOSITE CORE)

LOCATION - AFT SURFACE OF AFT SHIELD PLATE - REACTOR CENTERLINE
RADIUS (IN) 0.0 STATION (IN) -10.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

NEUTRONS/CM2, E GT 1.0MEV	2.42E 15	1.71	8.38E 16	1.71	8.43E 16	1.71	8.81E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.85E 16	1.71	6.39E 17	1.71	6.42E 17	1.71	6.71E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	1.48E 15	1.88	5.12E 16	1.88	5.15E 16	1.88	5.38E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	6.36E 06	1.09	2.27E 08	1.09	2.28E 08	1.09	2.38E 08	1.09

TABLE 7 - 4

MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE INTERNAL SHIELD
(GRAPHITE CORE)

LOCATION - AFT SURFACE OF AFT SHIELD PLATE - REACTOR CENTERLINE
RADIUS (IN) 0.0 STATION (IN) -10.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

NEUTRONS/CM2, E GT 1.0MEV	2.55E 15	1.59	8.84E 16	1.59	8.88E 16	1.59	9.29E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.06E 16	1.59	7.13E 17	1.59	7.16E 17	1.59	7.49E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	1.54E 15	1.73	5.31E 16	1.73	5.34E 16	1.73	5.58E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	6.06E 06	1.07	2.13E 08	1.07	2.14E 08	1.07	2.24E 08	1.07

TABLE 7-5

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION--TANGENT PLANE (MAX)
ALL BATH SHIELD (14.4" THICK)

$R = 0.0 \text{ cm}$
 $Z = 256.54 \text{ cm}$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	1.06(7)	1.48(7)	1.78(7)	1.8(7)	1.01
Fast Neutron Flux ($E > 1.0 \text{ MeV}$) ($\text{N}/\text{cm}^2\text{-sec}$)	4.52(11)	2.14(11)	6.63(11)	9.95(11)	2.0(12)	2.01
Int. Neutron Flux ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) ($\text{N}/\text{cm}^2\text{-sec}$)	1.13(12)	5.37(11)	1.66(12)	2.49(12)	3.0(12)	1.20
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) ($\text{N}/\text{cm}^2\text{-sec}$)	1.54(11)	7.30(10)	2.26(11)	3.39(11)	6.0(11)	1.77

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" aluminum end plates.

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105 10/17/70

TABLE 7-6
NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION--IPA

ALL BATH SHIELD (14.4" THICK)

[R = 91.44 cm
Z = 368.302 cm]

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	2.18(6)	3.05(6)	3.05(6)	7.0(6)	2.30
Fast Neutron Flux (E > 1.0 MeV) (N/cm ² -sec)	1.06(11)	3.49(10)	1.08(11)	1.08(11)	1.0(12)	9.36
Int. Neutron Flux (0.4 eV ≤ E ≤ 1.0 MeV) (N/cm ² -sec)	7.17(11)	2.36(11)	7.32(11)	7.32(11)	5.0(12)	6.85
Thermal Neutron Flux (E < 0.4 eV) (N/cm ² -sec)	1.27(11)	4.18(10)	1.30(11)	1.30(11)	1.0(12)	7.68

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data Location coordinates are referenced to CRAM model 0.0.

3) Shield Thickness includes 1.2" aluminum end plates.

138 10/22/70

111 10/22/70

717 9/24/70

TABLE 7-7

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION--LTS-1
ALL BATH SHIELD (14.4" THICK)

$R = 49.53 \text{ cm.}$
 $Z = 262.892 \text{ cm.}$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	5.90(6)	8.26(6)	1.24(7)	2.1(7)	1.70
Fast Neutron Flux ($E > 1.0 \text{ MeV}$) (N/cm ² -sec)	1.89(11)	1.09(11)	3.38(11)	6.76(11)	2.8(14)	414.0
Int. Neutron Flux ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) (N/cm ² -sec)	4.75(11)	2.73(11)	8.46(11)	1.69(12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) (N/cm ² -sec)	• 8.46(10)	4.87(10)	1.51(11)	3.02(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data location coordinates are referenced to CRAM model 0.0.

3) Shield thickness includes 1.2" end plates.

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111 10/22/70

TABLE 7-8
NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION--LTS-2
ALL BATH SHIELD (14.4" THICK)

[R = 49.53 cm.
Z = 267.972 cm.]

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	5.66(6)	7.92(6)	1.19(7)	1.8(7)	1.51
Fast Neutron Flux (E > 1.0 MeV) (N/cm ² -sec)	1.94(11)	1.06(11)	3.29(11)	6.58(11)	2.8(14)	425.0
Int. Neutron Flux (0.4 eV ≤ E ≤ 1.0 MeV) (N/cm ² -sec)	4.32(11)	2.36(11)	7.32(11)	1.46(12)	N/A	N/A
Thermal Neutron Flux (E < 0.4 eV) (N/cm ² -sec)	7.36(10)	4.01(10)	1.24(11)	2.48(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" end plates.

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111 10/22/70

TABLE 7-9

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION--LTS-3
ALL BATH SHIELD (14.4" THICK)

R = 49.53 cm.
Z = 290.832 cm.

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	4.66(6)	6.52(6)	9.78(6)	1.8(7)	1.84
Fast Neutron Flux ($E > 1.0$ MeV) (N/cm ² -sec)	1.31(11)	9.19(10)	2.84(11)	5.68(11)	2.8(14)	493.0
Int. Neutron Flux ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) (N/cm ² -sec)	2.84(11)	1.99(11)	6.17(11)	1.22(12)	N/A	N/A
Thermal Neutron Flux ($E \leq 0.4 \text{ eV}$) (N/cm ² -sec)	4.73(10)	3.31(10)	1.03(11)	2.06(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" end plates.

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111 10/22/70

TABLE 7-10

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION--TANGENT PLANE (MAX)
ALL BATH SHIELD (14.4" THICK)

R = 0.0 cm
Z = 256.54 cm

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr	---	1.17 (7)	1.64 (7)	1.97 (7)	1.8 (7)	0.914
Fast Neutron Flux ($E > 1.0$ MeV) (N/cm ² -sec)		2.19 (11)	6.79 (11)	1.02 (12)	2.0 (12)	1.96
Int. Neutron Flux ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) (N/cm ² -sec)		5.50 (11)	1.71 (12)	2.57 (12)	3.0 (12)	1.17
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) (N/cm ² -sec)		7.47 (10)	2.32 (11)	3.48 (11)	6.0 (11)	1.72

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" aluminum end plates.

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TABLE 7-11

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)DATA POINT LOCATION--TPA

R = 91.44 cm

ALL BATH SHIELD (14.4" THICK)

Z = 368.302 cm

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	2.37 (6)	3.32 (6)	3.32 (6)	7.0 (6)	2.11
Fast Neutron Flux (E > 1.0 MeV) (N/cm ² -sec)		3.60 (10)	1.12 (11)	1.12 (11)	1.0 (12)	8.93
Int. Neutron Flux (0.4 eV ≤ E ≤ 1.0 MeV) (N/cm ² -sec)		2.43 (11)	7.53 (11)	7.53 (11)	5.0 (12)	6.64
Thermal Neutron Flux (E < 0.4 eV) (N/cm ² -sec)		4.31 (10)	1.34 (11)	1.34 (11)	1.0 (12)	7.46

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data Location coordinates are referenced to CRAM model 0.0.
 3) Shield Thickness includes 1.2" aluminum end plates.

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TABLE 7-12
NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION--LTS-1
ALL BATH SHIELD (14.4" THICK)

[R = 49.53 cm.
Z = 262.892 cm.]

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	6.51 (6)	9.11 (6)	1.37 (7)	2.1 (7)	1.53
Fast Neutron Flux (E > 1.0 MeV) (N/cm ² -sec)		1.13 (11)	3.50 (11)	7.00 (11)	2.8 (14)	400.0
Int. Neutron Flux (0.4 eV ≤ E ≤ 1.0 MeV) (N/cm ² -sec)		2.83 (11)	8.77 (11)	1.75 (12)	N/A	N/A
Thermal Neutron Flux (E < 0.4 eV) (N/cm ² -sec)		5.05 (10)	1.57 (11)	3.14 (11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" end plates.

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TABLE 7-13

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION--LTS-2
ALL BATH SHIELD (14.4" THICK)

$R = 49.53 \text{ cm.}$
 $Z = 267.972 \text{ cm.}$

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	6.25 (6)	8.75 (6)	1.31 (7)	1.8 (7)	1.37
Fast Neutron Flux ($E > 1.0 \text{ MeV}$) ($\text{N}/\text{cm}^2\text{-sec}$)		1.09 (11)	3.38 (11)	6.76 (11)	2.8 (14)	414.0
Int. Neutron Flux ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) ($\text{N}/\text{cm}^2\text{-sec}$)		2.42 (11)	7.50 (11)	1.50 (12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) ($\text{N}/\text{cm}^2\text{-sec}$)		4.13 (10)	1.28 (11)	2.56 (11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" end plates.

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TABLE 7-14

NEUTRON AND PHOTON RADIATION ENVIRONMENT (GRAPHITE CORE)

DATA POINT LOCATION--LTS-3
ALL BATH SHIELD (14.4" THICK)

R = 49.53 cm.
Z = 290.832 cm.

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	5.16 (6)	7.22 (6)	1.08 (7)	1.8 (7)	1.67
Fast Neutron Flux (E > 1.0 MeV) N/cm ² -sec)		9.46 (10)	2.93 (11)	5.86 (11)	2.8 (14)	478.0
Int. Neutron Flux (0.4 eV $\leq$ E $\leq$ 1.0 MeV) (N/cm ² -sec)		2.05 (11)	6.36 (11)	1.27 (12)	N/A	N/A
Thermal Neutron Flux (E $\leq$ 0.4 eV) (N/cm ² -sec)		3.41 (10)	1.06 (11)	2.12 (11)	N/A	N/A

- NOTE: 1) Numbers in parentheses refer to powers of ten.
2) Data location coordinates are referenced to CRAM model 0.0.
3) Shield thickness includes 1.2" end plates.

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TABLE 7-15
NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION - TANGENT PLANE (MAX.)

[R = 0 cm.
 Z = 254.508 cm.]

SHIELD = 12.15" BATH, 0.25" LEAD

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	1.06(7)	1.49(7)	1.79(7)	1.8(7)	1.01
Fast Neutron Flux (E > 1.0 MeV) (N/cm ² -sec)	5.72(11)	2.76(11)	8.55(11)	1.28(12)	2.0(12)	1.56
Int. Neutron Flux (0.4 eV < E < 1.0 MeV) (N/cm ² -sec)	1.40(12)	7.15(11)	2.22(12)	3.33(12)	3.0(12)	0.90
Thermal Neutron Flux (E < 0.4 eV) (N/cm ² -sec)	2.12(11)	1.02(11)	3.17(11)	4.76(11)	6.0(11)	1.26

NOTE: 1) Numbers in parentheses refer to powers of ten.
 2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 7-16

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION - LTS-1

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 260.86 \text{ cm.} \end{bmatrix}$$

SHIELD = 12.15" BATH, 0.25" LEAD

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and 2nd Stream- ing Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr	---	5.82(6)	8.15(6)	1.22(7)	2.1(7)	1.72
Fast Neutron Flux ($E > 1.0$ MeV) (N/cm ² -sec)	2.44(11)	1.41(11)	4.37(11)	8.74(11)	2.8(14)	321.0
Int. Neutron Flux ($0.4 \text{ ev} < E < 1.0 \text{ MeV}$) (N/cm ² -sec)	6.40(11)	3.69(11)	1.14(12)	2.28(12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ ev}$) (N/cm ² -sec)	1.19(11)	6.87(10)	2.13(11)	4.26(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 7-17

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION - LTS-2

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 265.94 \text{ cm.} \end{bmatrix}$$

SHIELD = 12.15" BATH, 0.25" LEAD

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and 2nd Stream- ing Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	5.58(6)	7.81(6)	1.17(7)	1.8(7)	1.54
Fast Neutron Flux ($E > 1.0$ MeV) (N/cm ² -sec)	2.05(11)	1.36(11)	4.22(11)	8.44(11)	2.8(14)	332.0
Int. Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) (N/cm ² -sec)	5.82(11)	3.16(11)	9.80(1)	1.96(12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) (N/cm ² -sec)	1.04(11)	5.64(10)	1.74(11)	3.48(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 7-18

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION - LTS-3

$$\begin{bmatrix} R = 49.53 \text{ cm.} \\ Z = 288.80 \text{ cm.} \end{bmatrix}$$

SHIELD = 12.15" BATH, 0.25" LEAD

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1σ Uncertainty	KAP-V with 1σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	4.59(6)	6.42(6)	9.63(6)	1.8(7)	1.87
Fast Neutron Flux ($E > 1.0 \text{ MeV}$) (N/cm ² -sec)	1.69(11)	1.18(11)	3.66(11)	7.32(11)	2.8(14)	382.0
Int. Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) (N/cm ² -sec)	3.84(11)	2.98(11)	9.24(11)	1.85(12)	N/A	N/A
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) (N/cm ² -sec)	6.70(10)	4.67(10)	1.45(11)	2.90(11)	N/A	N/A

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data location coordinates are referenced to CRAM model 0.0.

TABLE 7-19

NEUTRON AND PHOTON RADIATION ENVIRONMENT (COMPOSITE CORE)

DATA POINT LOCATION - TPA

$$\begin{bmatrix} R = 51.44 \text{ cm.} \\ Z = 366.27 \text{ cm.} \end{bmatrix}$$

SHIELD = 12.15" BATH, 0.25" LEAD

	Nominal DOT-MAP Calculation	Nominal KAP-V Calculation	KAP-V with 1 σ Uncertainty	KAP-V with 1 σ Uncertainty and Streaming Correction	Specification	Ratio: Spec/KAP-V with 1 σ and Streaming
Photon Kerma Rate (Rad(C)/Hr)	---	2.17(6)	3.04(6)	3.04(6)	7.0(6)	2.30
Fast Neutron Flux ($E > 1.0$ MeV) (N/cm ² -sec)	1.18(11)	4.50(10)	1.38(11)	1.38(11)	1.0(12)	7.25
Int. Neutron Flux ($0.4 \text{ eV} < E < 1.0 \text{ MeV}$) (N/cm ² -sec)	7.45(11)	2.81(11)	8.71(11)	8.71(11)	5.0(12)	5.74
Thermal Neutron Flux ($E < 0.4 \text{ eV}$) (N/cm ² -sec)	1.32(11)	4.97(10)	1.54(11)	1.54(11)	1.0(12)	6.50

NOTE: 1) Numbers in parentheses refer to powers of ten.

2) Data location coordinates are referenced to CRAM model 0.0.

NOTE: Traverse at $z=190.58$ cm
on Figure 2-13.

Figure 7-1. Percent "Void" in Shield Regions Versus Radius

NOTE: Traverse at $z=190.58$ cm
on Figure 2-13.

Figure 7-2. Percent of Non-BATH Material in Shield Regions Versus Radius

Figure 7-3. Relative Radial Distribution of Heating in the Aft Central Shield Plate,
Region 35

(Composite Core)

Figure 7-4. Relative Axial Distribution of Heating in the Aft Central Shield Plate, Region 35 (Composite Core)

Figure 7-5. Relative Radial Distribution of Heating in the BATH Central Shield, Region 36
(Composite Core)

Figure 7-6. Relative Radial Distribution of Heating in the BATH Central Shield, Region 36
(Composite Core)

Figure 7-7. Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)

Figure 7-8. Relative Axial Distribution of Heating in the BATH Central Shield, Region 36 (Composite Core)

Figure 7-9. Relative Radial Distribution of Heating in the BATH Peripheral Shield I,
Region 38

(Composite Core)

Figure 7-10 Relative Radial Distribution of Heating in the BATH Peripheral Shield I,
Region 38

(Composite Core)

Figure 7-11. Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38 (Composite Core)

Figure 7-12. Relative Axial Distribution of Heating in the BATH Peripheral Shield I,
Region 38 (Composite Core)

Figure 7-13 Relative Radial Distribution of Heating in the BATH Peripheral Shield II,
Region 39

(Composite Core)

Figure 7-14 Relative Axial Distribution of Heating in the BATH Peripheral Shield II, Region 39 (Composite Core)

Figure 7-15 Relative Radial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Composite Core)

Figure 7-16 Relative Radial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Composite Core)

Figure 7-17. Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)

Figure 7-18. Relative Axial Distribution of Heating in the BATH Peripheral Shield III, Region 40 (Composite Core)

Figure 7-19. Relative Radial Distribution of Heating in the BATH Peripheral Shield IV,
Region 41

(Composite Core)

Figure 7-20 Relative Radial Distribution of Heating in the BATH Peripheral Shield IV, Region 41

(Composite Core)

Figure 7-21. Relative Axial Distribution of Heating in the BATH Peripheral Shield IV,
Region 41 (Composite Core)

Figure 7-22. Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41 (Composite Core)

Figure 7-23. Relative Radial Distribution of Heating in the Lead Central Shield, Region 43
(Composite Core)

Figure 7-24. Relative Radial Distribution of Heating in the Lead Central Shield, Region 43
(Composite Core)

Figure 7-25. Relative Axial Distribution of Heating in the Lead Central Shield, Region 43
(Composite Core)

Figure 7-26. Relative Axial Distribution of Heating in the Lead Central Shield, Region 43 (Composite Core)

Figure 7-27. Relative Radial Distribution of Heating in the Lead Peripheral Shield I,
Region 44

(Composite Core)

Figure 7-28. Relative Radial Distribution of Heating in the Lead Peripheral Shield I,
Region 44

(Composite Core)

Figure 7-29. Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)

Figure 7-30. Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44 (Composite Core)

Figure 7-31. Relative Radial Distribution of Heating in the Lead Peripheral Shield II,
Region 45

(Composite Core)

Figure 7-32. Relative Axial Distribution of Heating in the Lead Peripheral Shield II, Region 45 (Composite Core)

Figure 7-33 Relative Radial Distribution of Heating in the Lead Peripheral Shield III,
Region 46

(Composite Core)

Figure 7-34 Relative Radial Distribution of Heating in the Lead Peripheral Shield III,
Region 46

(Composite Core)

Figure 7-35. Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)

Figure 7-36. Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46 (Composite Core)

Figure 7-37. Relative Radial Distribution of Heating in the Lead Peripheral Shield IV,
Region 47

(Composite Core)

Figure 7-38 Relative Radial Distribution of Heating in the Lead Peripheral Shield IV,
Region 47

(Composite Core)

Figure 7-39. Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)

Figure 7-40. Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47 (Composite Core)

Figure 7-41. Relative Radial Distribution of Heating in the Peripheral Shield Plate,
Region 48

(Composite Core)

Figure 7-42. Relative Axial Distribution of Heating in the Peripheral Shield Plate, Region 48 (Composite Core)

Figure 7-43 Relative Radial Distribution of Heating in the Shield Plenum, Region 49
(Composite Core)

Figure 7-44. Relative Axial Distribution of Heating in the Shield Plenum, Region 49
(Composite Core)

Figure 7-45. Relative Radial Distribution of Photon Source in Region 15, Central Shield Plate - Aft

001-Z

Figure Z-46. Relative Axial Distribution of Photon Source in Region 15, Central Shield Plate - Aft

Figure 7-47. Relative Radial Distribution of Photon Source in Region 16, BATH Central Shield

Figure 7-48. Relative Axial Distribution of Photon Source in Region 16, BATH Central Shield

Figure 7-49. Relative Radial Distribution of Photon Source in Region 17, BATH Peripheral Shield - 1

Figure 7-5Q Relative Axial Distribution of Photon Source in Region 17, BATH Peripheral Shield-1

Figure 7-51. Relative Radial Distribution of Photon Source in Region 18, BATH Peripheral Shield - 2

Figure 7-52 Relative Axial Distribution of Photon Source in Region 18, BATH Peripheral Shield-2

Figure 7-53. Relative Radial Distribution of Photon Source in Region 19, BATH Peripheral Shield - 3

Figure 7-54 Relative Axial Distribution of Photon Source in Region 19, BATH Peripheral Shield-3

Figure 7-53. Relative Radial Distribution of Photon Source in Region 20, BATH Peripheral Shield - 4

Figure 7-56 Relative Axial Distribution of Photon Source in Region 20, BATH Peripheral Shield-4

Figure 7-57. Relative Radial Distribution of Photon Source in Region 21, Lead Central Shield

Figure 7-58. Relative Axial Distribution of Photon Source in Region 21, Lead Central Shield

Figure 7-59. Relative Radial Distribution of Photon Source in Region 22, Lead Peripheral Shield - 1

Figure 7-60. Relative Axial Distribution of Photon Source in Region 22, Lead Peripheral Shield - 1

Figure 7-61. Relative Radial Distribution of Photon Source in Region 23, Lead Peripheral Shield - 2

Figure 7-62. Relative Axial Distribution of Photon Source in Region 23, Lead Peripheral Shield - 2

Figure 7-63. Relative Radial Distribution of Photon Source in Region 24, Lead Peripheral Shield - 3

Figure 7-64. Relative Axial Distribution of Photon Source in Region 24, Lead Peripheral Shield - 3

Figure 7-65. Relative Radial Distribution of Photon Source in Region 25, Lead Peripheral Shield - 4

Figure 7-66. Relative Axial Distribution of Photon Source in Region 25, Lead Peripheral Shield - 4

Figure 7-67. Relative Radial Distribution of Photon Source in Region 26, Peripheral Shield Plate

Figure 7-68. Relative Axial Distribution of Photon Source in Region 26, Peripheral Shield Plate

Figure 7-69. Relative Radial Distribution of Photon Source in Region 27, Shield Plenum

Figure 7-70. Relative Axial Distribution of Photon Source in Region 27, Shield Plenum

Figure 7-71. Relative Radial Distribution of Heating in the Aft Central Shield Plate,
Region 35

(Graphite Core)

Figure 7-72 Relative Axial Distribution of Heating in the Aft Central Shield Plate, Region 35
(Graphite Core)

Figure 7-73 Relative Radial Distribution of Heating in the BATH Central Shield, Region 36
(Graphite Core)

Figure 7-74 Relative Axial Distribution of Heating in the BATH Central Shield, Region 36
(Graphite Core)

Figure 7-75 Relative Radial Distribution of Heating in the BATH Central Shield, Region 36
(Graphite Core)

Figure 7-76 Relative Axial Distribution of Heating in the BATH Central Shield, Region 36
(Graphite Core)

Figure 7-77. Relative Radial Distribution of Heating in the BATH Peripheral Shield I,
Region 38

(Graphite Core)

Figure 7-78 Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38
(Graphite Core)

Figure 7-79. Relative Radial Distribution of Heating in the BATH Peripheral Shield I,
Region 38

(Graphite Core)

Figure 7-80 Relative Axial Distribution of Heating in the BATH Peripheral Shield I, Region 38
(Graphite Core)

Figure 7-81. Relative Radial Distribution of Heating in the BATH Peripheral Shield II,
Region 39

(Graphite Core)

Figure 7-82 Relative Axial Distribution of Heating in the BATH Peripheral Shield II, Region 39
(Graphite Core)

Figure 7-83 Relative Radial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Graphite Core)

Figure 7-84 Relative Radial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Graphite Core)

Figure 7-85 Relative Axial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Graphite Core)

Figure 7-86 Relative Axial Distribution of Heating in the BATH Peripheral Shield III,
Region 40

(Graphite Core)

Figure 7-87. Relative Radial Distribution of Heating in the BATH Peripheral Shield IV,
Region 41

(Graphite Core)

Figure 7-88 Relative Radial Distribution of Heating in the BATH Peripheral Shield IV,
Region 41

(Graphite Core)

Figure 7-89 Relative Axial Distribution of Heating in the BATH Peripheral Shield IV,
Region 41

(Graphite Core)

Figure 7-90 Relative Axial Distribution of Heating in the BATH Peripheral Shield IV, Region 41
(Graphite Core)

Figure 7-91. Relative Radial Distribution of Heating in the Lead Central Shield, Region 43
(Graphite Core)

Figure 7-92 Relative Radial Distribution of Heating in the Lead Central Shield, Region 43
(Graphite Core)

Figure 7-93 Relative Axial Distribution of Heating in the Lead Central Shield, Region 43
(Graphite Core)

Figure 7-94 Relative Axial Distribution of Heating in the Lead Central Shield, Region 43
(Graphite Core)

Figure 7-95 Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44
(Graphite Core)

Figure 7-96 Relative Radial Distribution of Heating in the Lead Peripheral Shield I, Region 44
(Graphite Core)

Figure 7-97. Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44
(Graphite Core)

Figure 7-98 Relative Axial Distribution of Heating in the Lead Peripheral Shield I, Region 44
(Graphite Core)

Figure 7-99. Relative Radial Distribution of Heating in the Lead Peripheral Shield II, Region 45
(Graphite Core)

Figure 7-100 Relative Axial Distribution of Heating in the Lead Peripheral Shield II, Region 45
(Graphite Core)

Figure 7-101. Relative Radial Distribution of Heating in the Lead Peripheral Shield III,
Region 46

(Graphite Core)

Figure 7-102. Relative Radial Distribution of Heating in the Lead Peripheral Shield III, Region 46

(Graphite Core)

Figure 7-103 Relative Axial Distribution of Heating in the Lead Peripheral Shield III, Region 46
(Graphite Core)

Figure 7-104 Relative Axial Distribution of Heating in the Lead Peripheral Shield III,
Region 46

(Graphite Core)

Figure 7-105 Relative Radial Distribution of Heating in the Lead Peripheral Shield IV,
Region 47

(Graphite Core)

Figure 7-106 Relative Radial Distribution of Heating in the Lead Peripheral Shield IV,
Region 47

(Graphite Core)

Figure 7-107. Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47
(Graphite Core)

Figure 7-108 Relative Axial Distribution of Heating in the Lead Peripheral Shield IV, Region 47
(Graphite Core)

Figure 7-109. Relative Radial Distribution of Heating in the Peripheral Shield Plate, Region 48
(Graphite Core)

Figure 7-110. Relative Axial Distribution of Heating in the Peripheral Shield Plate, Region 48
(Graphite Core)

Figure 7-111. Relative Radial Distribution of Heating in the Shield Plenum, Region 49
(Graphite Core)

Figure 7-112 Relative Axial Distribution of Heating in the Shield Plenum, Region 49
(Graphite Core)

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Figure 7-113 Maximum Gamma Ray Kerma Rate on the Tangent Plane
Versus Centerline Thickness of BATH Material

Figure 7-14. Maximum Fast ($E > 1.0$ Mev) Flux on the Tangent Plane Versus Centerline Thickness of BATH Material

8.0 REFLECTOR ASSEMBLY

8.1 INTRODUCTION

The Reflector Assembly consists of a hollow right circular cylinder which encompasses the core from its forward to aft surface. The basic load-bearing member is an aluminum support structure to which are attached beryllium segments required to fulfill the primary function of neutron reflection. The reactivity control function is satisfied by 18 control drums which are mounted within the reflector and rotate on bearings mounted on each end of the reflector. Rotation of the drums changes the position of the control plates to vary the reactivity.

8.1.1 Description of Aluminum Structure Reflector

8.1.1.1 Structure

The primary support structure is the aluminum structural cylinder which is utilized to transmit the NSS axial and radial loads to the interfacing nozzle flange and ultimately to the aft pressure vessel flange. Additionally the structure must withstand the radial fluid pressure drop and support the reflector beryllium segments and control drums.

8.1.1.2 Reflector Segments

Beryllium segments, required to provide the necessary nuclear characteristics for neutron reflection, are attached to the aluminum structure. Short segments of beryllium are utilized and are attached by the lateral support spring capsules; hence the beryllium is not used as a part of the structural support system of the reflector assembly.

8.1.1.3 Control Drums

Eighteen control drums are located within the beryllium portion of the reflector assembly. The cylindrical beryllium drums have copper boron composite control vanes mounted within a recess at the outer diameter of the beryllium. The control drum is encased within a thin wall aluminum structure. Axial holes through the beryllium cylinder and a channel around the control vane provide for adequate removal of heat from the control drum assembly.

8.1.1.4 Lateral Support System

The Lateral Support System is a Trade Study and is discussed in S-054 TS782, Lateral Support.

8.1.2 Content of Chapter

This chapter provides radiation information related to the Reflector Assembly (RA) ECC. The radiation analysis herein was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. The radiation data for the RA in this chapter consists of the following information:

- 1) the maximum radiation environment in the RA,
- 2) the maximum heating rate in each component in the RA,
- 3) the heating rate distributions in the heating rate model (See Section 8.4.2) regions which comprise the RA,
- 4) a summary of the maximum effects of Test Cell C on the RA environment,
- 5) the photon source distribution in the CRAM model (see Section 8.4.2) regions which comprise the RA, and
- 6) the transient radiation environment in the RA at selected times during the reactor operating profile.

8.2 SUMMARY OF RESULTS AND CONCLUSIONS

This Section presents a summary of the results and conclusions regarding the radiation data obtained for the RA.

8.2.1 Composite Core

This section presents the results obtained for the RA with the NSS containing a composite fueled core.

8.2.1.1 Radiation Environment

The maximum radiation environment in the RA ECC for the NSS containing a composite fueled core occurs at inner radius of the lateral support on the core midplane ($R \approx 19.5$ in. and $z + 26.0$ inches) for the gamma ray dose rate, the fast

neutron flux, and the intermediate neutron flux; however, the maximum thermal neutron flux occurs in the RA at a radius of 23.25 inches on the core midplane. At the inner radius of the lateral support, the gamma ray dose rate is 8.10×10^9 Rads-C/hr and the fast neutron ($E > 1.0$ Mev) flux is 1.72×10^{15} n/cm² - sec. The corresponding dose and fluence for one hour of full power operation is, respectively, 8.10×10^9 Rads-C and 6.19×10^{18} n/cm². While the corresponding dose and fluence for 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161 is, respectively, 8.10×10^{10} Rads-C and 6.19×10^{19} n/cm². To a 95 percent confidence level, the one standard deviation uncertainty factors associated with this data is 1.14 for gamma rays and 1.16 for fast neutrons.

A comparison of the maximum radiation environment in the RA ECC for the NSS with a composite fueled core to the maximum radiation environment in the RA ECC for the NSS with a graphite fueled core indicates the following differences:

- 1) the maximum gamma ray dose rate is 18 percent higher in the RA for the NSS with a graphite fueled core,
- 2) the maximum fast neutron flux is 7 percent lower in the RA for the NSS with a graphite fueled core,
- 3) the maximum intermediate neutron flux is 2 percent lower in the RA for the NSS with a graphite fueled core, and
- 4) the maximum thermal neutron flux is 2 percent lower in the RA for the NSS with a graphite fueled core and occurs at a different radial position.

8.2.1.2 Radiation Heating Rates

A total, of 19.9 Mw of heat is deposited to the RA ECC heating rate mode regions for the NSS containing a composite fueled core. Approximately 11 percent more heat is deposited in these regions for the NSS containing a graphite fueled core.

The maximum heating rate occurs in the control vane and is 7.56×10^4 Btu/hr-in³. This heating rate for the NSS containing a graphite fueled core is approximately 6 percent higher. To a 95 percent confidence level, the one standard deviation uncertainty factor for the heating rate in the beryllium portions of the RA is 1.18.

8.2.1.3 Radiation Sources

The total photon source in the RA ECC photon source model regions for the NSS containing a composite fueled core is 3.02×10^{19} Mev/sec and is primarily due to the A-286 springs. The photon source strength in these regions for the NSS containing a graphite fueled core is 0.8 percent lower. The one standard deviation uncertainty factor for the RA photon sources is 1.14.

For the NSS containing a composite fueled core, the photon sources in the RA ECC contribute 6 percent to the dose rate on a ten-foot meridian ring at the reactor centerline and 13 percent to the dose rate on the same meridian ring at the core midplane. Therefore, use of A-286 in these regions should be minimized in order to obtain the minimum weight internal shield.

8.2.2 Graphite Core

This Section presents the results obtained for the RA with the NSS containing a graphite fueled core.

8.2.2.1 Radiation Environment

The maximum radiation environment in the RA ECC occurs at the inner radius of the lateral support on the core midplane ($r = 19.5$ inches and $z = 26.0$ inches) for the gamma ray dose rate, the fast neutron flux, and the intermediate neutron flux; however, the maximum thermal neutron flux occurs in the RA at a radius of 22.45 inches on the core midplane. At the inner radius of the lateral support, the gamma ray dose rate is 9.60×10^9 Rads-C/hr and the fast neutron flux is 1.59×10^{15} n/cm²-sec. The corresponding dose and fluence for one hour of full power operation is, respectively, 9.60×10^9 Rads-C and 5.72×10^{18} n/cm² while the corresponding dose and fluence for 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161 is, respectively, 9.60×10^{10} Rads-C and 5.72×10^{19} n/cm². To a 95 percent confidence level, the one standard deviation uncertainty factors associated

with this data is 1.12 for gamma rays and 1.13 for fast neutrons.

8.2.2.2 Radiation Heating Rates

A total of 22.1 Mw of heat is deposited in the RA ECC heating rate model regions for the NSS containing a graphite fueled core. The maximum heating rate occurs in the control vane and is 8.03×10^4 Btu/hr-in³. To a 95 percent confidence level, the one standard deviation uncertainty factor for the heating rate in the beryllium portion of the RA is 1.14.

8.2.2.3 Radiation Sources

The total photon source in the RA ECC photon source model regions for the NSS containing a graphite fueled core is 3.00×10^{19} Mev/sec and is primarily due to the A-286 springs. The one standard deviation uncertainty factor for the photon sources is 1.14.

For the NSS containing a graphite fueled core, the photon sources in the RA contribute 5 percent of the dose rate on a ten-foot meridian ring at the reactor centerline and 13 percent to the dose rate on the same meridian ring at the core midplane.

8.3 REQUIREMENTS

This Section provides RA requirements which pertain to radiation data.

8.3.1 Specification Identification

The current governing specification for the Reflector Assembly (RA) is Specification No. EC-677559.

8.3.2 Applicable Requirements

The applicable EC-677559 requirements for the RA which are either evaluated from a radiation standpoint or require radiation information to be evaluated are as follows:

"3.1.1.1.3.1.3 Thermal-Hydraulic Performance. - The RA shall channel the flow from the nozzle tube outlet to a plenum at the forward end of the reflector in accordance with the following requirements.

3. 1. 2. 1 Reliability. - The reliability allocation for the RA is 0. 999839.

3. 1. 2. 4. 2. 1 Nuclear Environment. - The RA shall be designed to withstand
the induced nuclear environment specified in Table 10. "

8.4 METHOD OF ANALYSIS (Design Practices)

This Section describes the methods and models employed in the radiation analysis of the RA.

8.4.1 Analytical Procedures

Section 2.4.1 described 1) the overall framework of nuclear, thermal, and structural analyses, 2) the relationship of radiation and shielding computations to this overall framework, and 3) the details of the procedure which was employed to obtain radiation and shielding results for the NSS and all of the ECCs in the NSS which includes the RA.

8.4.2 Analytical Models

Section 2.4.2 provides a detailed description of

- 1) the position at which radiation data at selected locations in the NSS are provided (Figure 2-6 and Table 2-3),
- 2) the computational model employed to perform two-dimensional, discrete ordinates transport calculations with the DOT code (Figures 2-7 through 2-10),
- 3) the radiation heating rate model employed to report heating rates (Figures 2-11 and 2-12), and
- 4) the CRAM model employed to report photon source distributions (Figures 2-13 and 2-14).

Table 2-4 provides a description of the regions contained in each model and indicates which model regions are associated with a particular ECC. Table 2-4, Figure 2-11 for a composite fueled core and Figure 2-12 for a graphite fueled core indicate that the following regions comprise the RA ECC in the heating rate model:

- 1) Region 17, Lateral Support,
- 2) Region 18, Structure,
- 3) Regions 19 through 21, Reflector,
- 4) Region 26, Aft Reflector Hardware,
- 5) Region 31, Forward Reflector Hardware I, and
- 6) Region 34, Forward Reflector Hardware II.

Table 2-4, Figure 2-13 for a composite fueled core, and Figure 2-14 for a graphite fueled core indicate that the following regions comprise the RA ECC in the CRAM, or photon source, model:

- 1) Region 4, Lateral Support,
- 2) Region 5, Aluminum Structure,
- 3) Region 6, Reflector,
- 4) Region 8, Aft Reflector Hardware, and
- 5) Region 12, Forward Reflector Hardware.

8.4.3 Status/Maturity of Input Data

Section 2.4.3 described the type of input information required to perform the radiation analyses of the NSS which, of course, includes the RA ECC. This section then described the status/maturity of the input data and the effect on radiation analyses.

The input to radiation analysis for the RA consisted of atom densities which were based on weights calculated from WANL Drawing 940J418. Subsequent to the time at which these weight calculations were performed, the drawing for the RA had the list of materials updated and the aft end nozzle-reflector seal modified. The revised RA is shown in WANL Drawing 939J733. The change to the nozzle-reflector seal, while significant, does not have an appreciable effect on radiation data because of the position in the NSS at which the change occurs.

8.4.4 Validity of Analysis

This Section describes the comparison of analytical and experimental results which were employed to determine the validity of the analytical procedure and the uncertainties in the radiation data obtained from this procedure.

8.4.4.1 Comparison with Experimental Results

Section 2.4.4.1 describes all of the experiments which were performed on the PAX reactor so that calculated and experimental results could be compared. Section 2.4.4.1 also provides the major conclusions that were reached for each experiment. Of the experiments performed, the Radiation Environment in Various Reactor Components Experiment ⁽¹⁾ provides results which applied to the RA ECC.

8.4.4.2 Uncertainty Analysis

The results for the analysis of statistical variations between calculated and experimental data for the Radiation Environment in Various Reactor Components Experiment are presented in Section 2.4.4.2. The derivation of uncertainty factors for radiation environment data (fast, intermediate, and thermal neutron fluxes and gamma ray dose rates) and radiation heating rates (both neutron and photon) via the procedures outlined in NRP 600 and NRP 605 are also described in Section 2.4.4.2.

Uncertainty factors are derived for possible variations in the radiation data at a point (pointwise uncertainty factor, F_p) and for the possible variations in average radiation data over a region (i.e., a regionwise uncertainty factor, F_r , for the total neutron heating rate in the reflector.) Table 2-18 summarizes and describes these uncertainty factors for the RA with the NSS containing a composite fueled core. Table 2-26 provides similar information for the RA ECC with the NSS containing a graphite fueled core.

8.5 RESULTS, ANALYSIS, AND INTERPRETATION

This Section presents the results of the radiation analysis of the RA and interpretation of the significance of these results.

8.5.1 Composite Core

The radiation analysis of the RA was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. This Section presents radiation results for the RA with the NSS containing a composite fueled core.

8.5.1.1 Steady State

This Section provides radiation data for the RA with the NSS at steady state conditions which are defined to be 1) a composite fueled core in the SOL condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at EOL.

8.5.1.1.1 Radiation Environment

Table 8-1 presents the maximum radiation environment in the RA ECC for the NSS containing a composite fueled core. The maximum environment in the RA ECC occurs at the inner radius of the lateral support on the core midplane ($r=19.5$ inches and $z=26.0$ inches) for the gamma ray dose rate, the fast neutron flux, and the intermediate neutron flux; however, the maximum thermal neutron flux occurs in the RA at a radius of 23.25 inches on the core midplane. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. Also presented in this table are the neutron fluences and gamma ray doses for one hour of full power (1511 Mw) reactor operation and 60 ten minute cycles of the reactor over the power profile defined by Figure 2-161.

Section 8.5.1.2.1 presents the maximum environment integrated over the nominal reactor operating profile at several state points during operation.

The radiation environment (fast, intermediate and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the RA are presented in Section 2.5.1.1.1. Section 2.5.1.2.1 presents radiation data at selected location in the RA integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides a list of major NSS materials and the safe radiation level which can occur before significant radiation damage occurs. By comparing these levels with the preceding results and the radiation to be presented in Section 8.5.2.1.1 for a graphite core, the effect of radiation damage in RA materials can be assessed.

8.5.1.1.2 Radiation Heating Rates

This Section provides the heating rate by region, heating rate distributions, and the effect of Test Cell C on radiation data for the RA with the NSS containing a composite fueled core.

8.5.1.1.2.1 Total Heating Rates

Table 2-49 provides the heating rates for the heating rate model regions (See Section 8.4.2) associated with the RA ECC for the NSS containing a composite fueled core.

This table indicates that 19.9 Mw of heat are deposited in the regions comprising the RA ECC. Section 2.5.1.1.4 describes data on this table in detail.

Table 8-2 provides the maximum heating rate in each component in the RA ECC.

This table provides the following information:

- 1) The drawing number for the RA ECC,
- 2) The item number of each component in the ECC as given by the drawing,
- 3) The part nomenclature,
- 4) The material of which the part is made, and
- 5) The maximum heating rate for the component (Btu/hr-in^3) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate occurs in the control vane (Item No. 16), and is 7.56×10^4 Btu/hr-in³ for the NSS containing a composite fueled core. The heating rate in the same component for the NSS containing a graphite fueled core is 6 percent higher.

8.5.1.1.2.2 Heating Rate Distributions

Figures 8-1 through 8-18 provide the relative radial and relative axial distribution of radiation heating for the heating rate model regions (See Section 8.4.2) which comprise the RA ECC. Figures 8-10 and 8-19 provide the relative axial and relative azimuthal distribution of radiation heating in the control vane. This data is provided for the NSS containing a composite fueled core.

The heating rate at any position within the RA regions is determined in the following manner:

- 1) Determine the value of the relative radial (relative azimuthal for the control vane) distribution of heating for the position of interest,
- 2) Determine the value of the relative axial distribution of heating for the position of interest,
- 3) Determine the average value of the heating rate from the appropriate curve (the average value of the heating rate appears on both the radial and axial distribution curves for each region), and
- 4) The heating rate at the position of interest is the product of the average heating rate, the value of the relative radial (relative azimuthal for the control vane) distribution, and the value of the relative axial distribution.

8.5.1.1.2.3 Effect of Test Cell C on Radiation Data

Section 2.5.1.1.5 presents the results, and a description of these results, which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data indicate that the maximum change in the RA radiation information occur near the pressure vessel side and can be quite substantial. However, with the exception of the thermal neutron flux, the change in the RA radiation data at the inner radius of the lateral support is generally less than 2 percent. At this location, the thermal neutron flux is 8 percent higher with Text Cell C present.

8.5.1.1.3 Radiation Sources

Table 2-36 provides gamma ray source strengths for each of the regions in the CRAM or photon source, model which comprise the RA ECC (See Section 8.4.2). Section 2.5.1.1.2 provides a description of the data presented in this table.

Figures 8-20 through 8-29 provide the relative radial and relative axial distribution of the photon source for the CRAM model regions which comprise the RA ECC. The data in Figures 8-20 through 8-29 represents the distributions in these regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occur with the change in cores.

The photon source strength at any position and for any given energy group in the RA ECC is determined in the following manner:

- 1) Determine the value of the relative radial distribution of photon source at the position of interest,
- 2) Determine the value of the relative axial distribution of photon source at the position of interest,
- 3) Determine the source strength for the energy group and region of interest from Table 2-36, and
- 4) The source strength for the energy group and position of interest is the product of the source strength, the value of the relative radial distribution, and the value of the relative axial distribution.

8.5.1.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section 2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

8.5.1.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 8-3 presents the maximum radiation data in the RA ECC for the NSS with a composite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 8-3 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes this radiation data in more detail, and Table 2-51 provides similar information at selected locations in the RA ECC.

8.5.1.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-49 to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-49.

8.5.2 Graphite Core

The radiation analysis of the RA was performed for the NSS containing a composite fueled core and for the NSS containing a graphite fueled core. This section presents radiation results for the RA with the NSS containing a graphite fueled core.

8.5.2.1 Steady State

This Section provides radiation data for the RA with the NSS at steady state conditions which are defined to be 1) a graphite fueled core in the SOL condition, 2) nuclear properties which exist at the end of 60 minutes of full power operation, and 3) thermal-hydraulic conditions which exist at EOL.

8.5.2.1.1 Radiation Environment

Table 8-1 presents the maximum radiation environment in the RA ECC for the NSS containing a graphite fueled core. The maximum environment in the RA ECC occurs at the inner radius of the lateral support on the core midplane ($r=19.5$ inches and $z=26.0$ inches) for the gamma ray dose rate, the fast neutron flux, and the intermediate neutron flux; however, the maximum thermal neutron flux occurs in the RA at a radius of 22.45 inches on the core midplane. This table presents the fast, intermediate, and thermal neutron fluxes, the heating rate (w/gm-SS) and the gamma ray dose rate. Also presented in this table are the neutron fluences and gamma ray doses for one hour of full power (1511 Mw) reactor operation and 60 ten minute cycles of the reactor over the power profile defined by Fig.2-161. Section 8.5.2.2.1 presents the maximum environment integrated over the nominal reactor operating profile at several state points during operation.

A comparison of the maximum radiation environment in the RA ECC for the NSS with a composite fueled core to the maximum radiation environment in the RA ECC for the NSS with a graphite fueled core indicates the following differences:

- 1) the maximum gamma ray dose rate is 18 percent higher in the RA for the NSS with a graphite fueled core,

- 2) the maximum fast neutron flux is 7 percent lower in the RA for the NSS with a graphite fueled core,
- 3) the maximum intermediate neutron flux is 2 percent lower in the RA for the NSS with a graphite fueled core, and
- 4) the maximum thermal neutron flux is 2 percent lower in the RA for the NSS with a graphite fueled core and occurs at a different radial position.

The radiation environment (fast, intermediate, and thermal neutron fluxes and photon dose rate) along axial and radial traverses through the RA are presented in Section 2.5.2.1.1. Section 2.5.2.2.1 presents radiation data at selected locations in the RA integrated over the reactor operating profile.

The primary purpose of the internal radiation environment data is to provide support to thermal analyses so that thermal properties at the proper irradiated conditions are obtained. In addition, the internal radiation environment data is employed to evaluate the effects of radiation damage on the reactor materials. The S-110 Data Item provides a list of major NSS materials and the safe radiation level which can occur before significant radiation damage occurs. By comparing these levels with the preceding results, the effect of radiation damage in RA materials can be assessed.

8.5.2.1.2 Radiation Heating Rates

This Section provides the heating rate by region, heating rate distributions, and the effect of Test Cell C on radiation data for the RA with the NSS containing a graphite fueled core.

8.5.2.1.2.1 Total Heating Rates

Table 2-65 provides the heating rates for the heating rate model regions (See Section 8.4.2) associated with the RA ECC for the NSS containing a graphite fueled core. This table indicates that 22.1 Mw of heat are deposited in the regions comprising the RA ECC which is 11 percent more heat deposited in the RA than was deposited for the RA with the NSS containing a composite fueled core. Section 2.5.2.1.4 describes data on this table in detail.

Table 8-2 provides the maximum heating rate in each component in the RA ECC.

This table provides the following information:

- 1) The drawing number for the RA ECC,
- 2) The item number of each component in the ECC as given by the drawing,
- 3) The part nomenclature,
- 4) The material of which the part is made, and
- 5) The maximum heating rate for the component (Btu/hr-in^3) for the NSS containing either a composite fueled core or a graphite fueled core.

This table indicates that the maximum heating rate occurs in the control vane (Item No 16), and is $8.03 \times 10^4 \text{ Btu/hr-in}^3$ for the NSS containing a graphite fueled core. The heating rate in the same component for the NSS containing a composite fueled core is 6 percent lower.

8.5.2.1.2.2 Heating Rate Distributions

Figures 8-30 through 8-47 provide the relative radial and relative axial distribution of radiation heating for the heating rate model regions (See Section 8.4.2) which comprise the RA ECC. Figures 8-39 and 8-48 provide the relative axial and the relative azimuthal distribution of radiation heating in the control vane. This data is provided for the NSS containing a graphite fueled core. Section 8.1.1.2.2 describes how the heating rate at any position within the RA is obtained from these data.

8.5.2.1.2.3 Effect of Test Cell C on Radiation Data

Section 2.5.2.1.5 presents the results, and a description of these results, which indicate the effect of Test Cell C on the radiation data internal to the NSS. The data presented represents the maximum effect of Test Cell C on the radiation data.

Briefly, these data indicate that the maximum change in the RA radiation information occur near the pressure vessel side and can be quite substantial. However, with the exception of the thermal neutron flux, the change in the RA radiation data at the inner radius of the lateral support is generally less than 2 percent. At this location the thermal neutron flux is 8 percent higher with Test Cell C present.

8.5.2.1.3 Radiation Sources

Table 2-55 provides gamma ray source strengths for each of the regions in the CRAM or photon source, model which comprise the RA ECC (See Section 8.4.2). Section 2.5.2.1.2 provides a description of the data presented in this table.

Figures 8-20 through 8-29 provide the relative radial and relative axial distribution of the photon source for the CRAM model regions which comprise the RA ECC. The data in Figures 8-20 through 8-29 represents the distributions in these regions for the NSS containing either a composite fueled core or a graphite fueled core because no change in the distributions occur with the change in cores. Section 8.5.1.1.3 describes how the photon source strength at any position and for any given energy group is obtained from this data.

8.5.2.2 Transient

Figure 2-161 displays the power profile, percent full power as a function of time, for the NSS which employs either a composite fueled core or a graphite fueled core. Section 2.5.1.2 describes this profile and how the transient data provided in this section relates to the State Points indicated on the profile.

8.5.2.2.1 Radiation Environment

Figures 2-162 through 2-164 provide the integrating factors to convert fluxes, dose rates and heating rates to fluences, doses, and heat. Section 2.5.1.2.1 describes the manner in which these integrating factors are applied to radiation data to obtain integrated data over the reactor operating profile.

Table 8-4 presents the maximum radiation data in the RA ECC for the NSS with a graphite fueled core integrated over the power profile to the point in time at which each State Point occurs.

Table 8-4 presents the following information for each State Point:

- 1) the gamma dose and associated uncertainty factor (See Section 2.4.4.2 for the details concerning the uncertainty factor),
- 2) the fast neutron fluence and associated uncertainty factor,
- 3) the intermediate neutron fluence and associated uncertainty factor, and
- 4) the thermal neutron fluence and associated uncertainty factor.

Section 2.5.1.2.1 describes this radiation data in more detail, and Table 2-66 provides similar information at selected locations in the RA ECC.

8.5.2.2.2 Heating Rates

Figure 2-165 presents the prompt total reactor power and the delayed total reactor power as a function of run time for the power profile defined in Figure 2-161. Figure 2-166 presents the power after shutdown due to delayed photons and beta particles as a function of time after shutdown for the NSS with either a graphite or a composite fueled core.

Figures 2-162 through 2-164 provided the integrating factors to convert the total heating rate for a region or material in a region, as supplied by Table 2-65 to the total heat deposition at any point in time during the run profile defined by Figure 2-161. It should be noted that the neutron, prompt, and delayed integrating factors must be applied to the corresponding heating rate in Table 2-65.

8.6 References

1. D. J. Hill and W. P. Rankin, "Radiation Environment in Various Reactor Components", WANL-TME-1914 (July, 1970)

TABLE 8-1

MAXIMUM RADIATION ENVIRONMENT AT REFLECTOR ASSEMBLY
COMPOSITE AND GRAPHITE FUEL IN THE REACTOR CORE

Description	Location		Gamma Environment			
	Radius (in.)	Reactor Station (in.)	Heating Rate (w/gm-SS)	Dose Rate (Rads-C/hr)	1 Hour Dose (Rads-C)	10 Hour Dose (Rads-C)
Composite Core						
Outer Radius Core Periphery - Core Midplane	19.476	26.000	2.31(1)	8.10(9)	8.10(9)	8.10(10)
Internal to Be Reflector - Core Midplane	23.250	26.000	---	---	---	---
Graphite Core						
Outer Radius Core Periphery - Core Midplane	19.476	26.000	2.89(1)	9.60(9)	9.60(9)	9.60(10)
Internal to Be Reflector - Core Midplane	22.450	26.000	---	---	---	---

Description	Neutron Environment								
	Fast ($E > 1.0$ MeV)			Intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)			Thermal ($E < 0.4$ eV)		
	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence (n/cm^2)	Flux ($n/cm^2\text{-sec}$)	1 Hour Fluence (n/cm^2)	10 Hour Fluence ($n/cm^2\text{-sec}$)
Composite Core									
Outer Radius Core Periphery - Core Midplane	1.72(15)	6.19(18)	6.19(19)	1.02(16)	3.67(19)	3.67(20)	---	---	---
Internal to Be Reflector - Core Midplane	---	---	---	---	---	---	5.93(14)	2.13(18)	2.13(19)
Graphite Core									
Outer Radius Core Periphery - Core Midplane	1.59(15)	5.72(18)	5.72(19)	9.97(15)	3.59(19)	3.59(20)	---	---	---
Internal to Be Reflector - Core Midplane	---	---	---	---	---	---	5.81(14)	2.09(18)	2.09(19)

(18)

TABLE 8-2

MAXIMUM HEATING RATE BY COMPONENT IN THE REFLECTOR, ALUMINUM STRUCTURE
(940J418)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
1	Structure	Al	2.54(3)	3.08(3)
2	Ring, Forward Extension	Al	9.48(2)	1.12(3)
3	Ring, Aft Support	Al	3.97(2)	4.70(2)
4	Segment, Inner Reflector	Be	2.39(3)	2.55(3)
5	Segment, Outer Reflector	Be	1.24(3)	1.34(3)
6	Can, Reflector Baffle	Al	6.73(2)	7.95(2)
7	Cone, Aft Baffle	Al	1.46(2)	1.73(2)
8	Liner, Drum Bore	Al	1.91(3)	2.29(3)
9	Cylinder, Drum	Be	2.25(3)	2.39(3)
10	Can, Drum	Al	1.82(3)	2.19(3)
11	Shaft, Forward Drum	Al	7.33(2)	8.44(2)
12	Shaft, Aft Drum	Al	4.04(2)	4.64(2)
13	Spring, Drum Cylinder	CRES	1.11(3)	1.23(3)
14	Housing, Forward BRG	Al	7.33(2)	8.44(2)
15	Support, Aft Bearing	Al	4.04(2)	4.64(2)
16	Vane, Control	CuB ¹⁰	7.56(4)	8.03(4)
17	STOP	CRES	1.43(3)	1.73(3)
18	Spring, Antibacklash	CRES	1.65(3)	1.85(3)
19	Coupling	CRES	1.61(3)	1.80(3)
20	Bearing	CRES	1.56(3)	1.80(3)
21	Standoff	CRES	7.86(2)	9.52(2)
22	Pad, Standoff	Al	1.78(2)	2.11(2)
23	Clamp, Aft Support Ring	Al	2.81(2)	3.47(2)

NOTE: Numbers in parentheses indicate powers of ten.

This table is continued on following page

TABLE 8-2 (Continued)
MAXIMUM HEATING RATE BY COMPONENT IN THE REFLECTOR, ALUMINUM STRUCTURE
(940J418)

Item Number	Part Nomenclature	Material	Max. Heating Rate (BTU/hr-in. ³)	
			Composite Fuel	Graphite Fuel
24	Capsule	Al	1.91(3)	2.29(3)
25	Cap, Capsule	Al	7.16(2)	8.60(2)
26	Sleeve	CRES	7.30(3)	8.85(3)
27	Spring, Segment Retainer	CRES	2.36(3)	2.80(3)
28	Plunger	Graphite	2.83(3)	3.22(3)
29	Spring, Lateral Support	CRES	4.13(3)	5.00(3)
30	Segment, Outer Seal	Graphite	3.18(3)	3.60(3)
31	Segment, Inner Seal	Graphite	3.80(3)	4.29(3)
32	Segment, Insulation	Grafoil	1.51(3)	1.71(3)
33	Liner, Insulation	Tungsten	9.86(3)	1.22(4)
34	Pin, Locating	Al	1.91(3)	2.29(3)
35	Nut, Adjustment	CRES	6.15(2)	7.43(2)
36	Seal, Aft Structure Outer	CRES	7.40(2)	9.00(2)
37	Seal, Aft Structure Inner	Al	3.01(2)	3.62(2)
38	Seal, Forward Structure	Al	6.43(2)	7.19(2)
39	Seal, Capsule	Al	1.91(3)	2.29(3)
40	Seal, Capsule Cap	Al	7.16(2)	8.60(2)
41	Seal, Forward Impedance	CRES	1.05(3)	1.20(3)
42	Insulator, Drum Cylinder	Grafoil	9.45(2)	1.07(3)
43	Retainer, Insulation	Al	1.78(3)	2.15(3)
44	Spring, Drive Shaft	CRES	1.36(3)	1.52(3)
45	Housing, Spring	Al	4.58(2)	5.11(2)
46	Filler, Segment	Al	7.20(2)	8.64(2)

NOTE: Numbers in parentheses indicate powers of ten.

TABLE 8 - 3
MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE REFLECTOR
(COMPOSITE CORE)

LOCATION -	OUTER RADIUS OF CORE PERIPHERY - CORE MIDPLANE								
	RADIUS (IN)	19.48	STATION (IN)	26.00					

					STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)	
					NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	
NEUTRONS/CM2, E GT 1.0MEV	2.62E 16	1.16	9.07E 17	1.16	9.11E 17	1.16	9.53E 17	1.16	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.64E 17	1.16	5.69E 18	1.16	5.71E 18	1.16	5.97E 18	1.16	
NEUTRONS/CM2, E LT 0.4EV	9.58E 15	1.35	3.31E 17	1.35	3.33E 17	1.35	3.48E 17	1.35	
GAMMA RAY DOSE, RADS (CARBON)	3.62E 07	1.14	1.45E 09	1.14	1.46E 09	1.14	1.53E 09	1.14	

TABLE 8 - 4
MAXIMUM TIME INTEGRATED RADIATION ENVIRONMENT IN THE REFLECTOR
(GRAPHITE CORE)

LOCATION -	OUTER RADIUS OF CORE PERIPHERY - CORE MIDPLANE								
RADIUS (IN)	19.48	STATION (IN)		26.00					

	STATE POINT (B)	STATE POINT (C)	STATE POINT (D)	STATE POINT (E)					
	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR	NOMINAL UNC FACTR					
NEUTRONS/CM2, E GT 1.0MEV	2.83E 16	1.13	9.81E 17	1.13	9.86E 17	1.13	1.03E 18	1.13	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.68E 17	1.13	5.82E 18	1.13	5.85E 18	1.13	6.11E 18	1.13	
NEUTRONS/CM2, E LT 0.4EV	9.77E 15	1.29	3.38E 17	1.29	3.40E 17	1.29	3.55E 17	1.29	
GAMMA RAY DOSE, RADS (CARBON)	3.13E 07	1.12	1.23E 09	1.12	1.24E 09	1.12	1.30E 09	1.12

Figure 8-1. Relative Radial Distribution of Heating in the Lateral Support, Region 17
(Composite Core)

Figure 8-2. Relative Axial Distribution of Heating in the Lateral Support,
Region 17 (Composite Core)

Figure 8-3. Relative Radial Distribution of Heating in the Structure, Region 18
(Composite Core)

Figure 8-4. Relative Radial Distribution of Heating in the Structure, Region 18
(Composite Core)

Figure 8-5. Relative Axial Distribution of Heating in the Structure, Region 18
(Composite Core)

Figure 8-6. Relative Axial Distribution of Heating in the Structure, Region 18
(Composite Core)

Figure 8-7. Relative Radial Distribution of Heating in the Reflector, Region 19 - 21
(Composite Core)

Figure 8-8. Relative Axial Distribution of Heating in the Reflector, Region 19 - 21
(Composite Core)

Figure 8-9. Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21
(Composite Core)

Figure 8-10. Relative Axial Distribution of Heating in the Reflector, Region 19 - 21
(Composite Core)

Figure 8-11. Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21
(Composite Core)

Figure 8-12. Relative Axial Distribution of Heating in the Reflector, Region 19 - 21
(Composite Core)

Figure 8-13 Relative Radial Distribution of Heating in the Aft Reflector Hardware,
Region 26

(Composite Core)

Figure 8-14. Relative Axial Distribution of Heating in the Aft Reflector Hardware,
Region 26 (Composite Core)

Figure 8-15. Relative Radial Distribution of Heating in the Forward Reflector Hardware I, Region 31

(Composite Core)

Figure 8-16. Relative Axial Distribution of Heating in the Forward Reflector Hardware I, Region 31 (Composite Core)

Figure 8-17. Relative Radial Distribution of Heating in the Forward Reflector Hardware II, Region 34

(Composite Core)

Figure 8-18. Relative Axial Distribution of Heating in the Forward Reflector Hardware II, Region 34 (Composite Core)

Figure 8-19. Azimuthal Heating Rate Distribution For The Control Drum Vanes With The Control Drums Positioned At An Angle of 50 Degrees

Figure 8-20. Relative Radial Distribution of Photon Source in Region 4, Lateral Support

Figure 8-21. Relative Axial Distribution of Photon Source in Region 4, Lateral Support

Figure 8-22 Relative Radial Distribution of Photon Source in Region 5, Aluminum Structure

Figure 8-23. Relative Axial Distribution of Photon Source in Region 5, Aluminum Structure

Figure 8-24. Relative Radial Distribution of Photon Source in Region 6, Reflector

Figure 8-25. Relative Axial Distribution of Photon Source in Region 6, Reflector

Figure 8-26. Relative Radial Distribution of Photon Source in Region 8, Aft Reflector Hardware

Figure 8-27. Relative Axial Distribution of Photon Source in Region 8, Aft Reflector Hardware

Figure 8-28. Relative Radial Distribution of Photon Source in Region 12, Forward Reflector Hardware

Figure 8-29. Relative Axial Distribution of Photon Source in Region 12, Forward Reflector Hardware

Figure 8-30. Relative Radial Distribution of Heating in the Lateral Support, Region 17
(Graphite Core)

Figure 8-31. Relative Axial Distribution of Heating in the Lateral Support, Region 17
(Graphite Core)

Figure 8-32. Relative Radial Distribution of Heating in the Aluminum Structure, Region 18
(Graphite Core)

Figure 8-33. Relative Axial Distribution of Heating in the Structure, Region 18
(Graphite Core)

Figure 8-34. Relative Radial Distribution of Heating in the Aluminum Structure, Region 18
(Graphite Core)

Figure 8-35. Relative Axial Distribution of Heating in the Structure, Region 18
(Graphite Core)

Figure 8-36. Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-37. Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-38. Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-39. Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-40. Relative Radial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-41. Relative Axial Distribution of Heating in the Reflector, Regions 19 - 21
(Graphite Core)

Figure 8-42. Relative Radial Distribution of Heating in the Aft Reflector Hardware, Region 26
(Graphite Core)

Figure 8-43. Relative Axial Distribution of Heating in the Aft Reflector Hardware, Region 26 (Graphite Core)

Figure 8-44. Relative Radial Distribution of Heating in the Forward Reflector Hardware,
Region 31

(Graphite Core)

Figure 8-45. Relative Axial Distribution of Heating in the Forward Reflector Hardware I, Region 31

(Graphite Core)

Figure 8-46. Relative Radial Distribution of Heating in the Forward Reflector Hardware II,
Region 34

(Graphite Core)

Figure 8-47. Relative Axial Distribution of Heating in the Forward Reflector Hardware II, Region 34

(Graphite Core)

Figure 8-48. Azimuthal Heating Rate Distribution For The Control Drum Vanes With The Control Drums Positioned At An Angle of 50 Degrees

9.0 CONTROL DRUM DRIVE ASSEMBLY

9.1 INTRODUCTION

The Control Drum Drive Assembly (CDDA) shall consist of eighteen (18) individual subassemblies, each consisting of an electrical stepper motor actuator mechanism, a controller, and an actuator malfunction override device.

9.1.1 Description of Control Drum Drive Assembly

The Control Drum Drive Assembly (CDDA) shown in Reference 1 consists of eighteen (18) individual Control Drum Drive Units (CDDU) each consisting of 1) an actuator mechanism, 2) an Actuator Malfunction Over-ride Device (A-MOD), and, 3) a controller.

The actuator mechanism is an electric actuator, which in response to signals from the controller, provides a torque output used to impart rotary motion to a reactor control drum. The internal actuator components comprise three major subassemblies: (1) the primary power train which consists of an electric motor with a permanent magnet rotor and a transmission, (2) a shaft position monitoring output device consisting of two redundant potentiometers, and (3) the A-MOD trigger mechanism consisting of an electrical coil, permanent magnet, and a spring and hammer device. The actuators are mounted in a partially recessed configuration in the pressure vessel forward dome closure by means of pedestals which are machined as an integral part of the dome. The protruding portion of the actuator is enclosed in a pressure vessel housing. The forward end of this housing incorporates the base of the electrical connector. The large majority of these components are fabricated with CRES material.

The A-MOD, under normal operating conditions, is the drive shaft which transmits the torque output of the actuator mechanism to the control drum. For emergency operation, the A-MOD, upon a signal from the controls system, disengages the actuator mechanism and rotates the control drum to the "orifice angle" position from any position within the

working range of the drum. The actuator torque is transmitted to the NSS control drums through a tubular drive shaft, fabricated with 304 stainless steel, that passes through the forward dome closure and the internal peripheral shield. On the aft end of the drive shaft is a spline, fabricated with A-286, which is coupled to the control drum. The A-MOD drive mechanism consists of two major subassemblies: (1) the drive spring, fabricated with A-286, and located in the center of the drive shaft, and (2) the shaft position cam mechanism, fabricated with A-286 and tungsten carbide, which is mounted in the actuator pressure vessel forward dome closure mounting pedestal.

Control of the actuator is accomplished with a remotely located (position, undefined but will be removed from the vessel) electronic controller package, which in turn, responds to electrical signals generated by the EPIC.

The centerline of an actuator shaft is located 24.5 inches from the core centerline. The aft end of the coupling spline is the component of this ECC that is closest to the core. The aft position of the spline is at approximately station -3.6 inches.

9.1.2 Content of Chapter

This chapter provides the maximum nominal radiation environment for the shielded graphite and composite cores and an unshielded core at the CDDA subassemblies of the actuator, A-MOD and controller. The maximum nominal data are the calculated radiation environment at the subassembly position that yields the maximum radiation environment. These data do not include the application of the uncertainty factors and are thus designated as nominal. The data are provided for steady state and transient operation of the NSS.

The neutron environment data provided for steady-state operation of the NSS at 1511 MW thermal power are the neutron flux plus one- and ten-hour fluence for 1) fast neutrons ($E > 1.0$ MeV), 2) intermediate neutrons ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$), and, 3) thermal neutrons ($E < 0.4 \text{ eV}$). The photon environment data provided for steady state operation are

the photon dose rate (Rad-C/Hr) plus one- and ten-hour dose levels (Rad-C) and the heating rate (watt/gm) for selected materials. The transient data presented in this chapter provide similar time integrated information for the power profile presented in Figure 2-161. The uncertainties are also presented for these nominal data.

The estimate of the radiation environment at the actuator location is also provided due to the presence of Test Cell C for operation of the NSS in Test Cell C. These data are the heating rate (watts/gm) in stainless steel and the photon dose rate (Rad-C/Hr) and dose levels (Rad-C) for one- and ten-hour operation.

The photon source data from the A-MOD subassembly are not presented in this chapter, but are presented in Chapters 6.0 and 7.0. The actuators and controllers were not in the geometrical model of the computer problems, thus source data are not available for these subassemblies.

Additional information supplied are related to the following:

1. The requirements of the CDDA related to the nuclear environment.
2. A brief description of the analytical model, technique, and validity of techniques employed to calculate the radiation environment for the CDDA ECC.
3. The status/maturity of the data and presentation of the results and analysis. This includes a presentation of the uncertainties of the radiation environment and heating rates.

9.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section summarizes the radiation environment results of the Radiation and Shielding analysis performed on the shielded graphite and composite cores, an unshielded core and operation in Test Cell C. The results are presented according to fast neutron ($E > 1.0$ MeV), intermediate neutron ($0.4 \text{ eV} \leq E \leq 1.0$ MeV), and thermal neutron ($E < 0.4$ eV), environment plus heating rate and gamma dose data at the actuator, A-MOD and controller locations. The flux and dose data presented are after one-hour continuous operation at 1511 MW thermal power.

9.2.1 Fast Neutron ($E > 1.0$ MeV) Environment

The maximum nominal (the calculated maximum environment at the components without application of uncertainty factors) fast neutron fluxes at the actuator, controller, and A-MOD subassemblies for the shielded composite core are 2.96×10^{11} (3.1)*, 7.65×10^{11} (3.1), and 9.96×10^{13} (1.16) $\text{n/cm}^2\text{-sec}$, respectively. The corresponding ten-hour fluences are 1.07×10^{16} (3.1), 2.75×10^{16} (3.1), and 3.59×10^{18} (1.16) n/cm^2 . This is compared to the shielded graphite core fast neutron fluxes of 2.88×10^{11} (3.1), 7.00×10^{11} (3.1), and 9.50×10^{13} (1.13) $\text{n/cm}^2\text{-sec}$. at the actuator, controller and A-MOD, respectively. The corresponding ten-hour fluences are 1.04×10^{16} (3.1), 2.52×10^{16} (3.1), and 3.42×10^{18} (1.16) n/cm^2 .

The composite core fast neutron fluxes and fluences are slightly greater (3 to 9 percent) than those for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core.

The CDDA was investigated for operation with the internal shield removed from the NSS. The maximum nominal unshielded fast neutron flux levels for a graphite or composite core are 9.00×10^{12} (5.0), and 2.59×10^{13} (5.0) $\text{n/cm}^2\text{-sec}$ at the actuator and controller locations, respectively. The maximum nominal fast neutron fluxes at the A-MOD location are 9.50×10^{13} (1.13) and 9.96×10^{13} (1.16) $\text{n/cm}^2\text{-sec}$ for the graphite and composite cores, respectively. The corresponding ten-hour fluences (in sequence of fluxes) are 3.24×10^{17} (5.0), 9.32×10^{17} (5.0), 3.42×10^{18} (1.13), and 3.59×10^{18} (1.16) n/cm^2 .

The preceding data were presented for operation in space. The fast neutron flux at the actuator and controller locations is estimated to increase by a factor of 1.28 (ANISN calculations) due to the presence of Test Cell C. Thus the fast neutron flux environment is 1.15×10^{13} (5.0) and 3.32×10^{13} (5.0) $\text{n/cm}^2\text{-sec}$ for the actuator and

* Numbers in () in Section 9.2 denote the one sigma uncertainty factor that apply to the nominal data.

controller. These data are for unshielded operation in Test Cell C. The corresponding ten-hour fluences are 4.14×10^{17} (5.0) and 1.12×10^{18} (5.0) n/cm^2 . Since the location where the maximum occurs for the A-MOD is very close to the core, the perturbation due to Test Cell C is expected to be small. Further calculations are required to assess the magnitude.

9.2.2 Intermediate Neutron ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) Environment

The maximum nominal intermediate neutron flux at the actuator, controller, and A-MOD subassemblies for the shielded composite core are 6.49×10^{11} (3.1), 2.20×10^{12} (3.1), and 8.28×10^{14} (1.16) $\text{n/cm}^2\text{-sec}$, respectively. The corresponding ten-hour fluences are 2.34×10^{16} (3.1), 7.92×10^{16} (3.1), and 2.98×10^{19} (1.16) n/cm^2 . This is compared to the shielded graphite core fast neutron fluxes of 6.12×10^{11} (3.1), 2.10×10^{12} (3.1) and 7.62×10^{14} (1.13) $\text{n/cm}^2\text{-sec}$ at the actuator, controller and A-MOD, respectively. The corresponding ten-hour fluences are 2.20×10^{16} (3.1), 7.65×10^{16} (3.1), and 2.74×10^{19} (1.13) n/cm^2 .

The composite core intermediate neutron fluxes and fluences are slightly greater (5 to 9 percent) than those for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core.

The maximum nominal unshielded intermediate neutron flux levels for a graphite or composite core are 1.45×10^{14} (5.0) and 4.15×10^{14} (5.0) $\text{n/cm}^2\text{-sec}$ at the actuator and controller locations, respectively. The maximum nominal intermediate neutron fluxes at the A-MOD locations are 7.62×10^{14} (1.13) and 8.28×10^{14} (1.16) $\text{n/cm}^2\text{-sec}$ for the graphite and composite cores, respectively. The corresponding ten-hour fluences (in sequence of fluxes) are 5.22×10^{18} (5.0), 1.49×10^{19} (5.0), 2.74×10^{18} (1.13) and 2.98×10^{19} (1.16) n/cm^2 .

The preceding data were presented for operation in space. The intermediate neutron flux at the actuator and controller locations is estimated to increase by a factor of 1.78 (ANISN calculations) due to the presence of Test Cell C. Thus, the intermediate neutron flux environment is 2.58×10^{14} (5.0) and 7.39×10^{14} (5.0) $\text{n/cm}^2\text{-sec}$ for the actuator and controller. These data are for unshielded operation in Test Cell C. The corresponding ten-hour fluences are 9.28×10^{18} (5.0) and 2.66×10^{19} (5.0) n/cm^2 . Since the location where the maximum occurs for the A-MOD is very close to the core, the perturbation due to Test Cell C is expected to be small. Further calculations are required to assess the magnitude.

9.2.3 Thermal Neutron ($E < 0.4$ eV) Environment

The maximum nominal thermal neutron flux at the actuator, controller, and A-MOD subassemblies for the shielded composite core are 1.13×10^{11} (3.1), 3.65×10^{11} (3.1), and 1.29×10^{14} (1.35) $\text{n/cm}^2\text{-sec}$, respectively. The corresponding ten-hour fluences are 4.07×10^{15} (3.1), 1.31×10^{16} (3.1), and 4.64×10^{18} (1.35) n/cm^2 . This is compared to the shielded graphite core thermal neutron fluxes of 1.06×10^{11} (3.1), 3.45×10^{11} (3.1), and 1.27×10^{14} (1.29) $\text{n/cm}^2\text{-sec}$ at the actuator, controller, and A-MOD, respectively. The corresponding ten-hour fluences are 3.82×10^{15} (3.1), 1.24×10^{16} (3.1), and 4.57×10^{18} (1.29) n/cm^2 .

The composite core thermal neutron fluxes and fluences are slightly greater (2 to 7 percent) than those for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core.

The maximum nominal unshielded thermal neutron flux levels for a graphite or composite core are 6.03×10^{13} (5.0) and 1.73×10^{14} (5.0) $\text{n/cm}^2\text{-sec}$ at the actuator and controller locations, respectively. The maximum nominal thermal neutron fluxes at the A-MOD location are 1.27×10^{14} (1.29) and 1.29×10^{14} (1.35) $\text{n/cm}^2\text{-sec}$ for the graphite and composite cores, respectively. The

corresponding ten-hour fluences (in sequence of fluxes) are 2.17×10^{18} (5.0), 6.22×10^{18} (5.0), 4.57×10^{18} (1.29) and 4.64×10^{18} (1.35) n/cm^2 .

The preceding data were presented for operation in space. The thermal neutron flux at the actuator and controller locations is estimated to increase by a factor of 2.26 (ANISN calculations) due to the presence of Test Cell C. Thus, the thermal neutron flux environment is 1.36×10^{14} (5.0) and 3.91×10^{14} (5.0) n/cm^2 -sec at the actuator and controller locations, respectively. These data are for unshielded operation in Test Cell C. The corresponding ten-hour fluences are 4.90×10^{18} (5.0) and 1.41×10^{19} (5.0) n/cm^2 . Since the location where the maximum occurs for the A-MOD is very close to the core, the perturbation due to Test Cell C is expected to be small. Further calculations are required to assess the magnitude.

9.2.4 Gamma Dose Environment

The maximum nominal gamma dose (frequently referred to as Kerma) rates at the actuator, controller, and A-MOD subassemblies for the shielded composite core are 2.78×10^6 (1.4), 4.30×10^6 (1.4), and 9.69×10^8 (1.14) Rad-C/Hr, respectively. The corresponding ten-hour dose levels are 2.78×10^7 (1.4), 4.30×10^7 (1.4), and 9.69×10^9 (1.14) Rad-C. This is compared to the shielded graphite core gamma dose rates of 2.95×10^6 (1.4), 4.80×10^6 (1.4), and 1.10×10^9 (1.12) Rad-C/Hr at the actuator, controller, and A-MOD, respectively. The corresponding ten-hour dose levels are 2.95×10^7 (1.4), 4.80×10^7 (1.4), and 1.10×10^{10} (1.12) Rad-C.

The graphite core dose levels are slightly greater (4 to 12 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

The maximum nominal unshielded gamma dose rates for a graphite or composite core are 1.31×10^8 (2.0), and 2.91×10^8 (2.0) Rad-C/Hr at the actuator and controller locations, respectively. The maximum nominal gamma dose rates at the A-MOD location

are 1.10×10^9 (1.12) and 9.69×10^8 (1.14) Rad-C/Hr for the graphite and composite cores, respectively. The corresponding ten-hour dose levels, (in sequence of dose rates) are 1.31×10^9 (2.0), 2.91×10^9 (2.0), 1.10×10^{10} (1.12), and 9.69×10^9 (1.14) Rad-C.

The preceding data were presented for operation in space. The gamma dose rates at the actuator and controller locations due to the influence of Test Cell C is estimated to be 9.64×10^7 (2.0) Rad-C/Hr. Thus, the gamma dose rate environment is 2.27×10^8 (2.0) and 3.87×10^8 (2.0) Rad-C/Hr at the actuator and controller locations, respectively. The corresponding ten-hour dose levels are 2.27×10^9 (2.0) and 3.87×10^9 (2.0) Rad-C. These data are the sum of the estimated Test Cell C influence and the unshielded data for space operation. Since the location where the maximum occurs for the A-MOD is very close to the core, the perturbation due to Test Cell C is expected to be small. Further calculations are required to assess the magnitude.

9.2.5 Radiation Heating Environment

The maximum nominal heating rates at the actuator, controller, and A-MOD subassemblies for the shielded composite core are 8.91×10^{-3} (1.4), 1.56×10^{-2} (1.4), and 3.37 (1.14) watts per gram of stainless steel 304, respectively.

There are other materials in the actuator, e.g., brass, alnico V, polyimide, etc. The photon heating rate for materials with a low atomic number (except H) can be assumed to be the same as stainless steel on a watts/gm basis as a first approximation. For materials with high atomic number, e.g., Pt and Sm being considered for use in the permanent magnet, the heating rate can be assumed as equivalent to the tungsten heating rate on a watts/gm basis as a first approximation. The tungsten heating rate at the actuator location is 1.38×10^{-2} (1.4) watt/gm.

The maximum nominal heating rate at the actuator, controller, and A-MOD subassemblies for the shielded graphite core are 9.27×10^{-3} (1.4), 1.62×10^{-2} (1.4), and 3.78 (1.12) watts per gram of stainless steel 304. The tungsten heating rate at the actuator is 1.45×10^{-2} (1.4) watts/gm.

The graphite core heating rates are slightly greater (4 to 15 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

The maximum nominal unshielded heating rates for stainless steel 304 for a graphite or composite core are 4.37×10^{-1} (2.0) and 9.35×10^{-1} (2.0) watt/gm at the actuator and controller locations, respectively. The maximum nominal heating rates at the A-MOD location are 3.78 (1.12) and 3.37 (1.14) watt/gm (SS) for the graphite and composite cores, respectively.

The preceding data were presented for operation in space. The heating rates at the actuator and controller locations due to the influence of Test Cell C is estimated to be 3.07×10^{-1} (2.0) watts/gm (SS). Thus, the heating rate environment is 7.44×10^{-1} (2.0) and 1.24 (2.0) watts/gm (SS) at the actuator and controller locations, respectively. These data are the sum of the estimated Test Cell C influence and the unshielded data for space operation. Since the location where the maximum occurs for the A-MOD is very close to the core, the perturbations due to Test Cell C is expected to be small. Further calculations are required to assess the magnitude.

9.3 REQUIREMENTS

This section identifies the governing specification and the applicable requirements in the specification of the CDDA which require radiation information.

9.3.1 Specification Identification

The governing specification for the CDDA ECC is EC-677585.

9.3.2 Applicable Requirements

"3.1.2.1 Reliability. - The reliability allocation for the CDDA is 0.99982. In meeting this allocation, failure of one actuator-controller unit is allowable,

provided that it is accompanied by a successful AMOD actuator override and provides that there is no structural failure of the actuator mounting at its pressure vessel interface or of the AMOD during its normal operating function as a drive shaft.

3.1.2.4.2 Induced Environment.

3.1.2.4.2.1 Nuclear Environment. - The CDDA shall be designed to withstand the induced nuclear environment specified in Table 1.

3.1.2.4.2.3 Thermal Environment. - The CDDA shall be designed to withstand the thermal environment which is attendant to the thermal-hydraulic data specified in Table 5 (tbd). "

9.4 METHOD OF ANALYSIS - DESIGN PRACTICES

A brief description of the analytical model, analytical procedure, status/maturity of the input data and the validity of the analysis are presented in this section.

9.4.1 Analytical Procedures

The analytical procedure used to generate the radiation environment presented for the CDDA for the shielded and unshielded graphite and composite cores is presented in Section 2.4.2. Also presented in that section are the analytical procedures used to obtain the estimates of the Test Cell C induced environment. The gamma environment were obtained with the technique using the NRX-A6 test data and the neutron data were obtained using the I-D ANISN calculations.

9.4.2 Analytical Models

The only subassembly of the CDDA that was considered in the geometrical model used to obtain the radiation environment presented in this chapter was the A-MOD. The actuator and controller were not included in this model. The reasons for this are that: 1) the controller position and composition are not known, and 2) the exact representation of the actuators is not possible in the (R, Z) geometry employed in the DOT problems. A

smeared mass annulus simulating the actuators on a finer mesh dome and using higher order quadrature than that used in the present calculations could be performed, but scheduler problems did not permit time for this complex analysis.

The A-MOD material internal to the dome was the only material used in the analytical model. The representation of this subassembly was also simplified. The mass of the coupling spline and thrust collar was considered; however, the axial distribution of material due to the drive shaft tube and A-MOD drive spring was considered to be uniformly distributed from the top of the coupling spline to the dome closure. The masses of the coolant filter spring, piston ring, piston and adapter were not included in the model. The fact that the A-MOD drive spring was considered to terminate at the dome closure rather than at the piston compensates for the smaller mass components not considered.

9.4.3 Status/Maturity of Input Data

The status/maturity of the primary data required to perform the radiation level computations at the CDDA locations for the graphite and composite cores is presented in Section 2.4.3.

The status/maturity of other requirements is as follows:

1. The actuator was not present in the calculational models employed to obtain the radiation environment.
2. The controller design and location is unknown at this time. The location of the controller was assumed to be near the forward dome closure on the reactor centerline where the maximum radiation environment could possibly occur for this component.
3. The A-MOD material internal to the pressure vessel dome closure was assumed to be of axially uniform composition.
4. The gamma radiation data presented for the environment due to the presence of Test Cell C are based upon calculations performed on an unshielded 939J685E reactor with the application of approximations based upon engineering principles (discussed in Section 2.4.2).

Since the calculations are not rigorous, the data reflect larger uncertainties. Additional analysis is required to more accurately determine these data.

5. The neutron radiation data presented for Test Cell C are based upon one-dimensional ANISN calculations. Additional analysis is required to more accurately determine these data.

9.4.4 Validity of the Analysis

9.4.4.1 Comparison with Experimental Results

A discussion of the comparison of the calculated radiation environment with applicable experimental data is presented in Section 2.4.

9.4.4.2 Uncertainty Analysis

Table 9-5 through 9-7 included the uncertainty factors associated with the radiation environment presented in Tables 9-1 to 9-3. The derivation of these factors is presented in Section 2.4. The exceptions are those for the Test Cell C environments (presented in Table 9-4) which are based upon past NRX test experience and engineering judgment. The uncertainty in the radiation environment at the CDDA locations is considered in the design of this ECC.

A detailed preliminary Failure Mode Analysis (FMA) for the CDDA ECC is given in Reference 2. This preliminary analysis will be updated. Various failure modes are listed and are due to various mechanisms, such as fatigue, creep, changes in material properties due to radiation, etc.

The reliability analysis and assessment of these failure modes⁽³⁾ uses as input the probabilities of successful operation of the various components of the CDDA operating in a specified phase for an identified failure mode. These probabilities of successful operation are generated using the uncertainty factors in the radiation environment presented in Tables 9-5 and 9-6. The various disciplines that analyze the performance of the NSS, such as, Thermal and Stress Analysis, use these uncertainties in radiation heating, and neutron

flux and fluence to determine the uncertainties in the reactor temperatures and changes in material properties due to radiation. These overall uncertainties are then used to assist in the determination of an overall probability of success of the components operating in the specified thermal and nuclear environment.

The CDDA will be designed using, in part, the uncertainty factors for the radiation environment in order that a 0.99982 probability of success of meeting the performance characteristics with the exception of the Emergency Mode Operational Requirements for the Useful Life of the CDDA.

9.5 RESULTS, ANALYSIS, AND INTERPRETATION

Data are presented for the composite and graphite cores for steady state and transient operation of the NSS. Also included are data for operation of an unshielded NSS with estimates of the influence of Test Cell C on the radiation environment. Radiation data are presented as fast ($E > 1.0$ MeV), intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV) and thermal ($E < 0.4 \text{ eV}$) neutron flux plus one- and ten-hour neutron fluences. In addition, heating rates (watts/gm) and photon dose rates (Rad-C/Hr) and dose levels for one- and ten-hour operation are presented.

Also included are data for operation of an unshielded NSS plus estimates of the influence of Test Cell C on the radiation environment. The analytical models, analytical procedure and the status/maturity of the input data used to obtain the results presented in this chapter are described in Sections 9.4.2, 9.4.1 and 9.4.3, respectively.

9.5.1 Composite Core

9.5.1.1 Steady State

9.5.1.1.1 Radiation Environment

9.5.1.1.1.1 Fluxes and Dose Rates

Table 9-1 presents the steady state neutron fluxes and dose rates after one-hour operation at 1511 MW thermal power operation in space for several CDDA locations. This

table provides the maximum nominal radiation environment at the controller A-MOD, and actuator locations for a shielded NSS. These data are applicable after one-hour continuous operation of the NSS at 1511 MW.

Table 9-5 presents the uncertainty factors associated with the data presented in Table 9-1. The derivation of these uncertainties is presented in Section 2.4.4 with a discussion of the application of these values to the reliability analysis of the CDDA. These data are to be used to ensure that the design of the CDDA will attain the 0.99982 probability of success for operation with the internal shield in position.

The composite core neutron fluxes are slightly greater (2 to 9 percent) than those for the graphite core. The converse is true for the photon data where the photon dose/rates for the graphite core are slightly greater (4 to 12 percent) than that for the composite core. The difference in neutron environment is attributed to a lower carbon to uranium ratio in the composite fuel resulting in a higher median energy of fission and harder spectrum in the core than that for the graphite core. The lower photon environment for the composite core is attributed to the higher density composite fuel resulting in a greater attenuation of the core gamma rays by this core.

Additional flux and dose rate data at other CDDA locations can be obtained from the data presented in Section 2.5.1.

Section 2.5.1 presents ANISN calculated ratios of the increase in neutron fluxes due to the presence of Test Cell C at the actuator and controller locations. Applying these factors to the space operation data, yields the following respective actuator and controller nominal neutron environment for operation in Test Cell C: 1) 3.79×10^{11} and 9.80×10^{11} n/cm²-sec fast neutron flux, 2) 1.16×10^{12} and 3.91×10^{12} n/cm²-sec intermediate neutron flux, and, 3) 2.55×10^{11} and 8.24×10^{11} n/cm²-sec thermal neutron flux. The effect at the A-MOD location can not be determined with the present techniques, but the effect induced by Test Cell C is estimated to be small.

From the ratio of TCC to space gamma dose rate presented in Section 2.5.1, the gamma dose rate at the actuator and controller locations with operation in TCC is expected to be 1.39×10^7 and 2.15×10^7 Rad-C/Hr, respectively. This could be as large as 9.64×10^7 Rad-C/Hr which is an estimate based upon NRX-A6 test data related to unshielded operation in TCC. Further analysis is required to obtain a better estimate.

9.5.1.1.1.2 Fluence and Dose

Table 9-1 presents the maximum nominal steady state neutron fluences and dose levels for one- and ten-hour operation in space at 1511 MW. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma dose data are used to assist in the selection of materials that are capable of reliable performance in these environments. They are also used to determine fluence levels to which materials are to be irradiated in radiation effects tests of materials and to determine the change in material properties from radiation effects test data.

Using the flux and dose data presented in the previous section, the estimated ten-hour fluence and dose data at the respective actuator and controller for operation in Test Cell C is the following: 1) 1.37×10^{16} and 3.52×10^{16} n/cm² fast neutron, 2) 4.18×10^{16} and 1.41×10^{17} n/cm² intermediate neutron, 3) 9.18×10^{15} and 2.97×10^{16} n/cm² thermal neutron, and 4) 1.39×10^8 and 2.15×10^8 Rad-C gamma dose.

9.5.1.1.2 Radiation Heating Rates

Table 9-1 presents the maximum nominal heating rates at the CDDA subassembly locations. This table provides the following information related to the heating rates:

1. The heating rate (watts/gm) for stainless steel 304 and A-286 at the A-MOD.
2. The heating rate (watts/gm) for stainless steel 304 and tungsten at the actuator location.
3. The heating rate (watts/gm) for stainless steel 304 at an assumed controller location.

Additional heating rate data are provide in Section 2.5.1.1.

To obtain the heating rate in BTU/hr-in³ the watts/gm data are to be multiplied by 443, 449, and 1080 for stainless steel, A-286, and W, respectively. There are other materials in the actuator, e.g., brass, alnico V, polyimide, etc. The photon heating rate data for low atomic number (except H) can be assumed to be the same as stainless steel on a watts/gm basis as a first approximation. Data for materials with high atomic number, e.g., Pt and Sm being considered for use in the permanent magnet, can be assumed as equivalent to the tungsten heating rate on a watts/gm basis as a first approximation.

The heating rates for the graphite core are slightly greater (4 to 15 percent) than those for the composite core. This difference is attributed to the higher density composite fuel resulting in a greater attenuation of the core gamma rays.

The radiation heating induced by Test Cell C can be estimated from the ratio of the heating in Test Cell C to that in space. Application of this ratio yields a heating rate at the actuator and controller of 3.93×10^{-2} and 6.87×10^{-2} watts/gm (SS), respectively. This could be as large as 3.07×10^{-1} watts/gm (SS) which is an estimate based upon NRX-A6 test data related to unshielded operation in Test Cell C. Further analysis is required to obtain a better estimate.

9.5.1.2 Transient

Table 9-5 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the locations on the controller, actuator and A-MOD. Table 9-5 contains the following information:

1. The thermal neutron fluence ($E < 0.4$ eV).
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$).
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$).
4. The integral radiation heating (watt-sec/gm).
5. The gamma dose (Rad-C).

Table 9-5 also provides the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4.2 discusses the method by which these factors were obtained.

These transient data include the effect of fission product buildup for various operating times.

Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.1.2.

9.5.2 Graphite Core

9.5.2.1 Steady State

9.5.2.1.1 Radiation Environment

9.5.2.1.1.1 Fluxes and Dose Rates

Table 9-2 presents the steady state neutron fluxes and dose rates after one-hour operation at 1511 MW thermal power operation in space for several CDDA locations. This table provides the maximum nominal radiation environment at the controller A-MOD, and actuator locations for a shielded NSS.

Table 9-6 presents the uncertainty factors associated with the data presented in Table 9-2. The derivation of these uncertainties is presented in Section 2.4.4.2 with a discussion of the application of these values to the reliability analysis of the CDDA. These data are to be used to ensure that the design of the CDDA will attain the 0.99982 probability of success for operation with the internal shield in position.

The composite core neutron fluxes are slightly greater (2 to 9 percent) than those for the graphite core. The converse is true for the photon data where the photon dose rates for the graphite core are slightly greater (4 to 12 percent) than that for the composite core.

The difference in neutron environment is attributed to a lower carbon to uranium ratio in the composite fuel resulting in a higher median energy of fission and harder spectrum in the core than that for the graphite core. The lower photon environment for the composite core is attributed to the higher density composite fuel resulting in a greater attenuation of the core gamma rays.

Additional flux and dose rate data at other CDDA locations can be obtained from the data presented in Section 2.5.2.

Section 2.5.1 presents ANISN calculated ratios of the increase in neutron fluxes due to the presence of Test Cell C at the actuator and controller locations. Applying these factors to the space operation data, yields the following respective actuator and controller nominal neutron environment: 1) 3.69×10^{11} and 8.98×10^{11} $\text{n/cm}^2\text{-sec}$ fast neutron flux, 2) 1.09×10^{12} and 3.74×10^{12} $\text{n/cm}^2\text{-sec}$ intermediate neutron flux, and 3) 2.40×10^{11} and 7.80×10^{11} $\text{n/cm}^2\text{-sec}$ thermal neutron flux. The effect at the A-MOD location can not be determined with the present techniques, but the effect induced by Test Cell C is estimated to be small.

From the ratio of TCC to space gamma dose rate presented in Section 2.5.1, the gamma dose rate at the actuator and controller locations with operation in TCC is expected to be 1.48×10^7 and 2.40×10^7 Rac-C/Hr , respectively. This could be as large as 9.64×10^7 Rad-C/Hr which is an estimate based upon NRX-A6 test data related to unshielded operation in TCC. Further analysis is required to obtain a better estimate.

9.5.2.1.1.2 Fluence and Dose

Table 9-2 presents the maximum nominal steady state neutron fluences and dose levels for one- and ten-hour operation in space at 1511 MW. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma dose data are used to assist in the selection of materials that are capable of reliable performance in these environments. They are also

used to determine fluence levels to which materials are to be irradiated in radiation effects tests of materials and to determine the change in material properties from radiation effects test data.

Using the flux and dose data presented in the previous section, the estimated ten-hour fluence and dose data at the respective actuator and controller for operation in Test Cell C is the following: 1) 1.33×10^{16} and 3.23×10^{16} n/cm^2 fast neutron, 2) 3.92×10^{16} and 1.35×10^{17} n/cm^2 intermediate neutron, 3) 8.62×10^{15} and 2.80×10^{16} n/cm^2 thermal neutron, and 4) 1.48×10^8 and 2.40×10^8 Rad-C gamma dose.

9.5.2.1.2 Radiation Heating Rates

Table 9-2 presents the maximum nominal heating rates at the CDDA subassembly locations. This table provides the following information related to the heating rates:

1. The heating rate (watts/gm) for stainless steel 304 and A-286 at the A-MOD.
2. The heating rate (watts/gm) for stainless steel 304 and tungsten at the actuator location.
3. The heating rate (watts/gm) for stainless steel 304 at an assumed controller location.

Additional heating rate data are provided in Section 2.5.2.1.

To obtain the heating rate in BTU/hr-in^3 the watts/gm data are to be multiplied by 443, 449, and 1080 for stainless steel, A-286, and W, respectively. There are other materials in the actuator, e.g., brass, alnico V, polyimide, etc. The photon heating rate data for low atomic number (except H) can be assumed to be the same as stainless steel on a watts/gm basis as a first approximation. Data for materials with high atomic number, e.g., Pt and Sm being considered for use in the permanent magnet, can be assumed as equivalent to the tungsten heating rate on a watts/gm basis as a first approximation.

The heating rate for the graphite core are slightly greater (4 to 15 percent) than those for the composite core. This difference is attributed to the higher density composite fuel resulting in a greater attenuation of the core gamma rays.

The radiation heating induced by Test Cell C can be estimated from the ratio of the heating in Test Cell C to that in space. Application of this ratio yields a heating rate at the actuator and controller of 4.09×10^{-2} and 7.13×10^{-2} watts/gm (SS), respectively. This could be as large as 3.07×10^{-1} watts/gm (SS) which is an estimate based upon NRX-A6 test data related to unshielded operation in Test Cell C. Further analysis is required to obtain a better estimate.

9.5.2.2 Transient

Table 9-6 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the locations on the controller, actuator and A-MOD. Table 9-6 contains the following information:

1. The thermal neutron fluence ($E < 0.4$ eV).
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$).
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$).
4. The integral radiation heating (watt-sec/gm).
5. The gamma dose (Rad-C).

Table 9-6 also provides the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4.2 discusses the method by which these factors were obtained.

These transient data include the effect of fission product buildup for various operating times.

Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.2.2.

9.5.3 Unshielded Core

9.5.3.1 Radiation Environment

9.5.3.1.1 Steady State

Since the actuator is designed to operate with the internal shield removed, radiation data are provided in Table 9-3 for unshielded operation. Table 9-7 presents the uncertainty factors for operation in this mode. The data in Table 9-3 are for steady state operation at 1511 MW after one hour of continuous operation. In this analysis, the actuator was moved 13 inches closer to the core (Dwg. 940J420) when the internal shield was removed. It is observed that the removal of the internal shield plus moving 13 inches closer to the core increases the fast, thermal and intermediate neutron fluxes by factors of 31.2, 569, and 237, respectively at the actuator location. Similarly, the gamma dose rate is increased by a factor of 44.4. These are obtained as the ratio of the unshielded data to the graphite core data.

In order to provide estimates of the maximum flux and dose that the CDDA ECC would encounter, calculations of the radiation environment were performed for an unshielded NSS operating in Test Cell C. Table 9-4 presents the results of these calculations. Applying the neutron ratios presented in Section 2.5.1 to the space operation data yields the following respective actuator and controller nominal neutron environment for unshielded operation in Test Cell C: 1) 1.15×10^{13} and 3.32×10^{13} $\text{n/cm}^2\text{-sec}$ fast, 2) 2.58×10^{14} and 7.39×10^{14} $\text{n/cm}^2\text{-sec}$ intermediate, and 3) 1.36×10^{14} and 3.91×10^{14} $\text{n/cm}^2\text{-sec}$ thermal. These data yield ten-hour fluence levels at the respective actuator and controller locations of 1) 4.14×10^{17} and 1.12×10^{18} n/cm^2 fast, 2) 9.28×10^{18} and 2.66×10^{19} n/cm^2 intermediate, and, 3) 4.90×10^{18} and 1.41×10^{19} n/cm^2 thermal.

The gamma dose for unshielded operation in Test Cell C at the actuator and controller locations is 2.27×10^8 and 3.87×10^8 Rad-C/Hr, respectively. The corresponding ten-hour dose levels are 2.27×10^9 and 3.87×10^9 Rad-C.

The heating rate for unshielded operation in Test Cell C at the actuator and controller locations is 4.68×10^{-1} and 9.66×10^{-1} watts/gm (SS), respectively.

The influence of the radiation environment due to Test Cell C at the A-MOD location requires further analysis.

9.5.3.1.2 Transient

Table 9-7 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. These data were obtained by the methods outlined in Section 2.4. These data are provided at the CDDA locations. Table 9-7 provides the following information:

1. The thermal neutron fluence ($E < 0.4$ eV).
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0$ MeV).
3. The fast neutron fluence ($E > 1.0$ MeV).
4. The gamma dose (Rad-C).
5. The integral radiation heating (watt-sec/gm).

This table also includes the uncertainty factors associated with the data.

These transient data include the effect of fission product buildup for various operating times.

9.5.4 Activation Analysis

Estimations of the gamma dose rate due to neutron activation of a samarium-cobalt permanent magnet in a CDDA actuator were made as a function of time after shutdown following a 10 minute burn of an unshielded NSS. These calculations are overestimates since: 1) the self-shielding of the neutron and photon flux by the actuator was not considered, 2) the maximum flux at the actuator was used, and 3) for samarium, all the

gammas were assumed to be entitled for each decay. The after shutdown dose rate was compared with that due to the fission product gammas.⁽⁴⁾ The fission product decay data in Reference 4 for the 939J685E NSS were related to that expected for the 939J723 NSS by appropriately changing the dose rate to compensate for differences in the mass thickness of material in the forward end of the NSS.

The overestimates show that the gamma dose rate due to ^{60}Co and ^{153}Sm decay at 10 cm from the magnet is less than 10 percent of that due to decay gammas from the core for times less than 50 hours after shutdown. For distances greater than 10 cm from the magnet, the percentage becomes progressively less.

For an unshielded NSS, the dose due to the decay of ^{60}Co and ^{153}Sm is not significant when compared to that due to fission product decay.

9.6 REFERENCES

1. "Control Drum Actuator W/Malfunction Override Device, Layout of, " Drawing 939J990, September 30, 1970.
2. J. L. Burke, "Control Drum Drive Actuator/A-MOD Failure Mode Analysis, " DRM 52407, September 30, 1970. To be updated.
3. R. Schatz, "NSS Reliability Analysis Model-Preliminary R-202 Report in Support of Engine PDR, " DRM-53021, August 6, 1970. To be updated.
4. R. N. Nassano, S. L. Anderson and K. M. Furio, "Gamma Ray In-Flight Environment Around the R-1 (939J685E) Following Shutdown," DRM-51934, January 9, 1970.

TABLE 9-1
MAXIMUM NOMINAL RADIATION ENVIRONMENT AT SEVERAL CDDA LOCATIONS
FOR THE COMPOSITE CORE (DWG. 939J723)

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/Hr)	(Rad-C)	(Rad-C)
Actuator	24.5	-37.0	8.91(-3)	2.78(6)	2.78(6)	2.78(7)
Controller	0.0	-36.5	1.56(-2)	4.30(6)	4.30(6)	4.30(7)
A-MOD	23.6	-3.6	3.37(0)	9.69(8)	9.69(8)	9.69(9)

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV ≤ E ≤ 1.0 MeV			E > 1.0 MeV		
	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
Actuator	1.13(11)	4.07(14)	4.07(15)	6.49(11)	2.34(15)	2.34(16)	2.96(11)	1.07(15)	1.07(16)
Controller	3.65(11)	1.31(15)	1.31(16)	2.20(12)	7.92(15)	7.92(16)	7.65(11)	2.75(15)	2.75(16)
A-MOD	1.29(14)	4.64(17)	4.64(18)	8.28(14)	2.98(18)	2.98(19)	9.96(13)	3.59(17)	3.59(18)

NOTE: Numbers in parentheses refer to powers of ten, i. e., 8.91(-3) = 8.91 × 10⁻³. 572 10/08/70

645 10/30/70 577 10/08/70

239 10/05/70 441 10/12/70

Heating Rate at A-MOD for A-286 = 3.45 (0) watt/gm. 132 10/02/70

Heating Rate at Actuator for W = 1.38 (-2) watt/gm. 239 10/05/70

242 10/05/70 242 10/05/70

TABLE 9-2
MAXIMUM NOMINAL RADIATION ENVIRONMENT AT SEVERAL CDDA LOCATIONS
FOR THE GRAPHITE CORE (DWG. 939J723)

Location	Position		Gamma Environment			
	Radius (In.)	Station (In.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/Hr)	(Rad-C)	(Rad-C)
Actuator	24.5	-37.0	9.27(-3)	2.95(6)	2.95(6)	2.95(7)
Controller	0.0	-36.5	1.62(-2)	4.80(6)	4.80(6)	4.80(7)
A-MOD	23.6	-3.6	3.78(0)	1.10(9)	1.10(9)	1.10(10)

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV ≤ E ≤ 1.0 MeV			E > 1.0 MeV		
	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
Actuator	1.06(11)	3.82(14)	3.82(15)	6.12(11)	2.20(15)	2.20(16)	2.88(11)	1.04(15)	1.04(16)
Controller	3.45(11)	1.24(15)	1.24(16)	2.10(12)	7.65(15)	7.65(16)	7.00(11)	2.52(15)	2.52(16)
A-MOD	1.27(14)	4.57(17)	4.57(18)	7.62(14)	2.74(18)	2.74(19)	9.50(13)	3.42(17)	3.42(18)

NOTE: Numbers in parentheses refer to powers of ten, i. e., 9.27(-3) = 9.27 × 10⁻³.

Heating Rate at A-MOD for A-286 = 3.86 (0) watt/gm.

Heating Rate at Actuator for W = 1.45 (-2) watt/gm.

575	10/08/70
579	10/08/70
580	10/08/70
119	10/02/70
240	10/05/70
243	10/05/70
648	10/30/70

TABLE 9-3
MAXIMUM NOMINAL RADIATION ENVIRONMENT AT SEVERAL CDDA LOCATIONS
FOR OPERATION IN SPACE WITH THE INTERNAL SHIELD REMOVED

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/Hr)	(Rad-C)	(Rad-C)
Actuator	24.5	-24.0	4.37(-1)	1.31(8)	1.31(8)	1.31(9)
Controller	0.0	-24.0	9.35(-1)	2.91(8)	2.91(8)	2.91(9)
A-MOD	23.6	-3.6	3.78(0) 3.37(0)	1.10(9) 9.69(8)	1.10(9) 9.69(8)	1.10(10) 9.69(9)

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV ≤ E ≤ 1.0 MeV			E > 1.0 MeV		
	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)
Actuator	6.03(13)	2.17(17)	2.17(18)	1.45(14)	5.22(17)	5.22(18)	9.00(12)	3.24(16)	3.24(17)
Controller	1.73(14)	6.22(17)	6.22(18)	4.15(14)	1.49(18)	1.49(19)	2.59(13)	9.32(16)	9.32(17)
A-MOD	* 1.27(14) ** 1.29(14)	4.57(17) 4.64(17)	4.57(18) 4.64(18)	7.62(14) 8.28(14)	2.74(18) 2.98(18)	2.74(19) 2.98(19)	9.53(13) 9.96(13)	3.42(17) 3.59(17)	3.42(18) 3.59(18)

NOTES:

1) Data at A-MOD not influenced due to removal of internal shield.

2) Numbers in parentheses refer to powers of ten, i.e., 4.37(-1) = 4.37 x 10⁻¹

* Graphite Core
** Composite Core

403 11/05/70
429 11/05/70

TABLE 9-4
ESTIMATES OF THE RADIATION ENVIRONMENT AT THE CDDA ACTUATOR LOCATION
DUE TO THE PRESENCE OF TEST CELL C

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/Hr)	(Rad-C)	(Rad-C)
Actuator (Unshielded)	24.5	-24.0	3.07(-1)	9.64(7)	9.64(7)	9.64(8)

NOTE:

- 1) Data obtained from DRM-52369 for unshielded core and were scaled from 1575 MW to 1511 MW. As an estimate, these data are applicable to graphite and composite cores.
- 2) Estimates of Test Cell C influence on neutron data can be obtained from Section 2.5.
- 3) Numbers in parentheses refer to powers of ten, i.e., $3.07(-1) \equiv 3.07 \times 10^{-1}$.
- 4) Uncertainty factor is 2.0 for gamma ray data and 5.0 for neutron data.

TABLE 9-5

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL CONTROL DRUM DRIVE ASSEMBLY LOCATIONS FOR THE COMPOSITE FUELED CORE

LOCATION - ACTUATOR - COMPOSITE
RADIUS (IN) 24.50 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.48E-01	1.40	5.14E 00	1.40	5.16E 00	1.40	5.39E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.88E 12	3.10	1.69E 14	3.10	1.70E 14	3.10	1.77E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.07E 13	3.10	3.70E 14	3.10	3.72E 14	3.10	3.89E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	1.87E 12	3.10	6.46E 13	3.10	6.49E 13	3.10	6.78E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.28E 04	1.40	4.45E 05	1.40	4.47E 05	1.40	4.67E 05	1.40

LOCATION - CONTROLLER-COMPOSITE
RADIUS (IN) 0.0 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	2.59E-01	1.40	8.99E 00	1.40	9.04E 00	1.40	9.44E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.26E 13	3.10	4.36E 14	3.10	4.38E 14	3.10	4.58E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.63E 13	3.10	1.25E 15	3.10	1.26E 15	3.10	1.32E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	6.02E 12	3.10	2.08E 14	3.10	2.09E 14	3.10	2.19E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.98E 04	1.40	6.89E 05	1.40	6.92E 05	1.40	7.23E 05	1.40

TABLE 9-5 (Continued)

LOCATION - A-MOD-COMPOSITE									
RADIUS (IN)		23.60		STATION (IN)		-3.60			

TABLE 9-6

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL CONTROL DRUM DRIVE ASSEMBLY LOCATIONS FOR THE GRAPHITE FUELED CORE

LOCATION - ACTUATOR-GRAPHITE
RADIUS (IN) 24.50 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.53E-01	1.40	5.37E 00	1.40	5.40E 00	1.40	5.64E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.75E 12	3.10	1.64E 14	3.10	1.65E 14	3.10	1.73E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.01E 13	3.10	3.49E 14	3.10	3.51E 14	3.10	3.67E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	1.75E 12	3.10	6.04E 13	3.10	6.08E 13	3.10	6.35E 13	3.10
GAMMA RAY DOSE, RAD (CARBON)	1.33E 04	1.40	4.69E 05	1.40	4.71E 05	1.40	4.92E 05	1.40

LOCATION - CONTROLLER-GRAPHITE
RADIUS (IN) 0.0 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	2.64E-01	1.40	9.28E 00	1.40	9.33E 00	1.40	9.75E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.15E 13	3.10	3.99E 14	3.10	4.01E 14	3.10	4.19E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.46E 13	3.10	1.20E 15	3.10	1.21E 15	3.10	1.26E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	5.69E 12	3.10	1.97E 14	3.10	1.98E 14	3.10	2.07E 14	3.10
GAMMA RAY DOSE, RAD (CARBON)	2.18E 04	1.40	7.64E 05	1.40	7.68E 05	1.40	8.03E 05	1.40

TABLE 9-6 (Continued)

LOCATION - A-MOD-GRAPHITE
 RADIUS (IN) 23.60 STATION (IN) -3.60

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM- STAINLESS STEEL	5.84E 01	1.12	2.13E 03	1.12	2.14E 03	1.12	2.24E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.57E 15	1.13	5.42E 16	1.13	5.44E 16	1.13	5.69E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.26E 16	1.13	4.35E 17	1.13	4.37E 17	1.13	4.56E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.09E 15	1.29	7.24E 16	1.29	7.28E 16	1.29	7.61E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.72E 06	1.12	1.72E 08	1.12	1.73E 08	1.12	1.81E 08	1.12

TABLE 9-7

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL CONTROL DRUM DRIVE ASSEMBLY LOCATIONS FOR AN UNSHIELDED CORE

LOCATION - ACTUATOR - UNSHIELDED
 RADIUS (IN) 24.50 STATION (IN) -24.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	6.98E 00	2.00	2.49E 02	2.00	2.50E 02	2.00	2.62E 02	2.00
NEUTRONS/CM2, E GT 1.0MEV	1.48E 14	5.00	5.13E 15	5.00	5.16E 15	5.00	5.39E 15	5.00
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.39E 15	5.00	8.27E 16	5.00	8.31E 16	5.00	8.69E 16	5.00
NEUTRONS/CM2, E LT 0.4EV	9.94E 14	5.00	3.44E 16	5.00	3.46E 16	5.00	3.61E 16	5.00
GAMMA RAY DOSE, RADS (CARBON)	5.81E 05	2.00	2.07E 07	2.00	2.08E 07	2.00	2.18E 07	2.00

LOCATION - CONTROLLER - UNSHIELDED
 RADIUS (IN) 0.0 STATION (IN) -24.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.49E 01	2.00	5.33E 02	2.00	5.36E 02	2.00	5.60E 02	2.00
NEUTRONS/CM2, E GT 1.0MEV	4.27E 14	5.00	1.48E 16	5.00	1.48E 16	5.00	1.55E 16	5.00
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.84E 15	5.00	2.37E 17	5.00	2.38E 17	5.00	2.49E 17	5.00
NEUTRONS/CM2, E LT 0.4EV	2.85E 15	5.00	4.87E 16	5.00	9.92E 16	5.00	1.04E 17	5.00
GAMMA RAY DOSE, RADS (CARBON)	1.29E 06	2.00	4.61E 07	2.00	4.63E 07	2.00	4.84E 07	2.00

TABLE 9-7 (Continued)

LOCATION - A-MOD COMPOSITE CORE - UNSHIELDED
 RADIUS (IN) 23.60 STATION (IN) -3.60

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	5.32E 01	1.14	1.91E 03	1.14	1.92E 03	1.14	2.01E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.64E 15	1.16	5.68E 16	1.16	5.71E 16	1.16	5.97E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 16	1.16	4.72E 17	1.16	4.75E 17	1.16	4.96E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.13E 15	1.35	7.36E 16	1.35	7.39E 16	1.35	7.73E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	4.25E 06	1.14	1.53E 08	1.14	1.54E 08	1.14	1.61E 08	1.14

LOCATION - A-MOD GRAPHITE CORE - UNSHIELDED
 RADIUS (IN) 23.60 STATION (IN) -3.60

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	5.84E 01	1.12	2.13E 03	1.12	2.14E 03	1.12	2.24E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.57E 15	1.13	5.42E 16	1.13	5.44E 16	1.13	5.69E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.26E 16	1.13	4.35E 17	1.13	4.37E 17	1.13	4.56E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.09E 15	1.29	7.24E 16	1.29	7.28E 16	1.29	7.61E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.72E 06	1.12	1.72E 08	1.12	1.73E 08	1.12	1.81E 08	1.12

10.0 STRUCTURAL SUPPORT COOLANT ASSEMBLY

10.1 INTRODUCTION

The Structural Support Coolant Assembly (SSCA) shall consist of a valve module containing two actively redundant flow channels each consisting of an SSCV and SSBV with associated actuators, and interconnecting piping (SSCL) between the valve module and other engine components.

10.1.1 Description of Structural Support Coolant Assembly

The SSCA consists of 1) two structural support coolant valves and actuators, 2) two structural support block valves and actuators, and 3) one set of structural support coolant lines. Component (3) shall include such fittings, junctions, plena, bellows, and welds as needed to satisfy the SSCA performance. The two SSCA valves and actuators are located at different radii and stations and supply two actively redundant systems. The design of the structural support coolant valve actuators⁽¹⁾ is the same as the control drum drive actuator. A brief description of the actuator is provided in Section 9.0.

Analysis has been performed on the structural support coolant valve actuators only.

10.1.2 Content of Chapter

This chapter provides the nominal (uncertainty factors not applied) radiation environment for the shielded graphite and composite cores and an unshielded core at a selected location of each of the SSCV actuators. These two locations are at the axial distance from the core station zero to the centerline of the SSCV actuators at a radius near that of the permanent magnet of each SSCV actuator. The locations were scaled from ANSC Drawing 1137400, Revision C. The data are supplied for steady state and transient operation of the NSS.

The neutron environment data provided for steady state operation of the NSS at 1511 MW thermal power are the neutron flux plus one-and ten-hour fluence for 1) fast neutrons ($E > 1.0$ MeV), 2) intermediate neutrons ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$), and, 3)

thermal neutrons ($E < 0.4$ eV). The photon environment data provided for steady state operation are the photon dose rate (Rad-C/Hr) plus one-and ten-hour dose levels (Rad-C) and the heating rate (watt/gm) for Stainless Steel 304. The transient data presented in this chapter provide similar time integrated information for the power profile presented in Figure 2-161. The uncertainties in these data are also presented. These data are supplied for shielded and unshielded operation of the NSS.

The SSCV actuators were not in the geometrical model of the computer codes, thus photon source data are not available for this ECC.

Additional information supplied are related to the following:

1. The requirements of the SSCA when related to the nuclear environment.
2. A brief description of the analytical model, technique, and validity of techniques employed to calculate the radiation environment for the SSCA ECC.
3. The status/maturity of the data and presentation of the results and analysis. This includes a presentation of the uncertainties of the radiation environment and heating rates.

10.2 SUMMARY OF RESULTS AND CONCLUSIONS

This section summarizes radiation environment results of the Radiation and Shielding analysis performed on the shielded graphite and composite cores, and on an unshielded core. The results are presented according to fast neutron ($E > 1.0$ MeV), intermediate neutron ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$), and thermal neutron ($E < 0.4 \text{ eV}$), environment plus heating rate and gamma dose data at the two SSCV actuator locations. The flux and dose rate data presented are after one-hour continuous operation at 1511 MW thermal power.

The data locations for the two actuators are denoted as 1) SSCVA-1 which is at a radius of 49.0 inches and station of -103.0 inches, and 2) SSCVA-2 which is at a radius of 42.5 inches and station of -111.05. These locations are for the shielded data. The unshielded locations are moved 13 inches closer to the core when the internal shield is removed ⁽²⁾.

10.2.1 Fast Neutron ($E > 1.0$ MeV) Environment

The nominal fast neutron fluxes at the SSCVA-1 and SSCVA-2 locations for the shielded composite core are 2.71×10^{10} (3.1)* and 2.79×10^{10} (3.1) $\text{n/cm}^2\text{-sec}$, respectively. The corresponding ten-hour fluences are 9.76×10^{13} (3.1) and 1.00×10^{14} n/cm^2 . This is compared to the shielded graphite neutron fluxes of 2.79×10^{10} (3.1) and 2.88×10^{10} (3.1) $\text{n/cm}^2\text{-sec}$ at the SSCVA-1 and SSCVA-2, respectively with corresponding ten-hour fluences of 1.00×10^{15} (3.1) and 1.04×10^{15} (3.1) n/cm^2 .

The graphite core neutron fluxes are slightly greater (3 percent) than those for the composite core. This is the reverse effect observed at the CDDA locations where the composite core flux was greater than that for the graphite. This is attributed to differences in the analytical techniques. The SSCVA data were obtained with point kernel calculations and the CDDA were obtained from transport calculations. The differences in composite and graphite core data are so small that they can be considered to be the same. The nominal unshielded fluxes for the graphite or composite cores at the SSCVA-1 and SSCVA-2 locations are 2.35×10^{12} (5.0) and 2.51×10^{12} (5.0) $\text{n/cm}^2\text{-Sec}$, respectively. The corresponding ten-hour fluences are 8.45×10^{16} (5.0) and 9.04×10^{16} (5.0) n/cm^2 .

10.2.2 Intermediate Neutron ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) Environment

The nominal intermediate neutron fluxes at the SSCVA-1 and SSCVA-2 locations for the shielded composite core are 2.37×10^{11} (3.1) and 1.88×10^{11} (3.1) $\text{n/cm}^2\text{-sec}$, respectively. The corresponding ten-hour fluences are 8.53×10^{15} (3.1) and 6.77×10^{15} (3.1) n/cm^2 . This is compared to the shielded graphite neutron fluxes of 2.35×10^{11} (3.1) and 1.84×10^{11} (3.1) $\text{n/cm}^2\text{-sec}$ at the SSCVA-1 and SSCVA-2, respectively, with corresponding ten-hour fluences of 8.46×10^{15} (3.1) and 6.62×10^{15} (3.1) n/cm^2 .

The composite core fluxes are slightly greater (1 and 3 percent) than those for the graphite core. The converse was true for the fast neutron fluxes and is the same trend observed for the CDDA data. This is due to the application of the transport calculated

* Numbers in () in Section 10.2 denote the one sigma uncertainty factor that apply to the nominal data.

ratio of intermediate to fast fluxes to the point kernel calculated fast flux. The transport calculated fast and intermediate fluxes external to the NSS were greater for the composite than graphite cores. This causes the composite intermediate fluxes to be greater than those for the graphite cores at the SSCVA locations. The differences between the graphite and composite core fluxes are so small the fluxes may be considered to be the same.

The nominal unshielded fluxes for the graphite or composite cores at the SSCVA-1 and SSCVA-2 locations are 3.79×10^{13} (5.0) and 4.05×10^{13} (5.0) n/cm² - sec, respectively. The corresponding ten-hour fluences are 1.36×10^{18} (5.0) and 1.46×10^{18} (5.0) n/cm².

10.2.3 Thermal Neutron (E < 0.4 eV) Environment

The nominal thermal neutron fluxes at the SSCVA-1 and SSCVA-2 locations for the shielded composite core are 4.32×10^{10} (3.1) and 3.42×10^{10} (3.1) n/cm² - sec, respectively. The corresponding ten-hour fluences are 1.56×10^{15} (3.1) and 1.22×10^{15} (3.1) n/cm². This is compared to the shielded graphite neutron fluxes of 4.52×10^{10} (3.1) and 3.57×10^{10} (3.1) n/cm² - sec at the SSCVA-1 and SSCVA-2, respectively, with corresponding ten-hour fluences of 1.63×10^{15} (3.1) and 1.29×10^{15} (3.1) n/cm².

The graphite core fluxes are slightly greater (5 percent) than those for the composite core. This is the same trend as those for the fast neutron flux and the reverse of that obtained at the CDDA locations where the composite was slightly higher than the graphite. This difference at the SSCVA and CDDA comparisons is due to the use of a thermal to fast neutron ratio obtained from transport calculations. The difference in the ratio for the composite and graphite cores was close to one percent, thus the thermal fluxes reflect the difference due to the point kernel calculated fast neutron fluxes at the SSCVA locations. The differences in the composite and graphite cores is so small the fluxes may be considered to be the same. The nominal unshielded fluxes for the graphite or composite cores at the SSCVA-1 and SSCVA-2 locations are 1.57×10^{13} (5.0) and 1.68×10^{13} (5.0) n/cm² - sec, respectively. The corresponding ten-hour fluences are 5.65×10^{17} (5.0) and 6.04×10^{17} (5.0) n/cm².

10.2.4 Gamma Dose Environment

The nominal gamma dose (frequently referred to as Kerma) rate at the SSCVA-1

and SSCVA-2 locations for the shielded composite core are 1.72×10^6 (1.4) and 9.72×10^5 (1.4) Rad-C/hr, respectively. The corresponding ten-hour dose levels are 1.72×10^7 (1.4) and 9.72×10^5 (1.4) Rad-C. This is compared to the shielded graphite gamma dose rates of 1.80×10^6 (1.4) and 1.03×10^5 (1.4) Rad-C/Hr at the SSCVA-1 and SSCVA-2, respectively, with corresponding ten-hour dose levels of 1.80×10^7 (1.4) and 1.03×10^7 (1.4) Rad-c.

The graphite core dose levels are slightly greater (5 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

The nominal unshielded gamma dose rates for the graphite or composite cores are 2.92×10^7 (2.0) and 2.86×10^7 (2.0) Rad-c/Hr at the SSCVA-1 and SSCVA-2 locations, respectively. The corresponding ten hour fluences are 2.92×10^8 (2.0) and 2.86×10^8 (2.0) Rad-c.

10.2.5 Radiation Heating Environment

The nominal heating rates at the SSCVA-1 and SSCVA-2 locations for the shielded composite core are 6.24×10^{-3} (1.4) and 3.53×10^{-3} (1.4) watts per gram of stainless steel (SS) 304, respectively. This is compared to the shielded graphite core heating rates of 6.51×10^{-3} (1.4) and 3.72×10^{-3} (1.4) watts/gm (SS) at the SSCVA-1 and SSCVA-2 locations, respectively.

The graphite core heating rates are slightly greater (4 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

The nominal unshielded heating rates for the graphite or composite cores are 9.52×10^{-2} (2.0) and 9.29×10^{-2} (2.0) watts/gm (SS) at the SSCVA-1 and SSCVA-2 locations, respectively.

10.3 REQUIREMENTS

This section presents governing specifications and the applicable requirements in the specification of the SSCA which require radiation information.

10.3.1 Specification Identification

The governing specification for the SSCA ECC is EC-677575

10.3.2 Applicable Requirements

3. 1. 2. 1 Reliability. - The reliability allocation for the SSCA is 0.999740.

3. 1. 2. 4. 2 Induced Environment.

3. 1. 2. 4. 2. 1 Nuclear Environment. - The SSCA shall be designed to withstand the induced nuclear environment shown in Table 10.

3. 1. 2. 4. 2. 3 Thermal Environment. - The SSCA shall be designed to withstand the thermal environment attendant to the thermal-hydraulic data specified in Tables 1 through 9, 13, and 14. "

*Numbers preceding the titles are the paragraph numbers in Specification EC-677575.

**Table number refers to that in EC-677575.

10.4 METHOD OF ANALYSIS - DESIGN PRACTICES

A brief description of the analytical model, analytical procedure, status/maturity of the input data and the validity of the analysis are discussed in this section.

10.4.1 Analytical Procedures

The analytical procedure used to generate the radiation environment for the shielded and unshielded graphite and composite cores is presented in Section 2.4.2.

10.4.2 Analytical Model

The analytical model used to describe the NSS is presented in Section 2.4.1. The SSCA was not included in the geometrical model since all of the components are external to the pressure vessel.

10.4.3 Status/Maturity of Input Data

The status/maturity of the primary data required to perform the radiation level computations at the SSCA locations for the graphite and composite cores is presented in Section 2.4.3.

The status/maturity of other input data is as follows:

1. There were not any components of the SSCA used in the geometry model of the calculations.
2. The spatial distribution of the radiation environment within the volume enclosed by the SSCA was not calculated.
3. Data are not presented for operation in Test Cell C due to a lack of definition of the location of these in the Test Cell.

10.4.4 Validity of the Analysis

10.4.4.1 Comparison With Experimental Results

A discussion of the comparison of the calculated radiation environment with the applicable experimental data is presented in Section 2.4.

10.4.4.2 Uncertainty Analysis

Tables 10-4 through 10-6 include the uncertainty factors associated with the radiation environment presented in Tables 10-1 to 10-3. The derivation of these factors is presented in Section 2.4. The uncertainty in the radiation environment at the SSCA locations is considered in the design of this ECC.

A detailed preliminary Failure Mode Analysis (FMA) for the SSCA ECC is given in Reference 3. This preliminary analysis will be updated. Various failure modes are listed and are due to various mechanisms, such as, fatigue, creep, changes in material properties due to radiation, etc.

The reliability analysis and assessment of these failure modes⁽⁴⁾ uses as input the probabilities of successful operation of the various components of the SSCA operating in a specified phase for an identified failure mode. These probabilities of successful operation are generated using the uncertainty factors in the radiation environment presented in Tables 10-4 and 10-5. The various disciplines that analyze the performance of the NSS, such as, Thermal and Stress Analysis, use these uncertainties in radiation heating, and neutron flux and fluence to determine the uncertainties in the reactor temperatures and changes in material properties due to radiation. These overall uncertainties are then used to assist in the determination of an overall probability of success of the components operating in the specified thermal and nuclear environment.

The SSCA will be designed using, in part, the uncertainty factors for the radiation environment in order that a 0.999900 probability of success of meeting the performance characteristics with the exception of the Emergency Mode Operational Requirements for the Useful Life of the SSCA.

10.5 RESULTS, ANALYSIS, AND INTERPRETATION

Data are presented for the composite and graphite cores for steady state and transient operation of the NSS. Also included are data for operation of an unshielded NSS. Radiation data are presented as fast ($E > 1.0$ MeV), intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)

and thermal ($E < 0.4 \text{ eV}$) neutron flux plus one- and ten-hour neutron fluences. In addition, heating rates (watts/gm) and photon dose rates (Rad-c/Hr) and dose levels for one- and ten-hour operation are presented. The analytical models, analytical procedure and the status/maturity of the input data used to obtain the results presented in this chapter are discussed in Sections 10.4.2, 10.4.1 and 10.4.3, respectively.

10.5.1 Composite Core

10.5.1.1 Steady State

10.5.1.1.1 Radiation Environment

10.5.1.1.1.1 Fluxes and Dose Rates

Table 10-1 presents the steady state neutron fluxes and dose rates after one-hour operation at 1511 MW thermal power in space for several SSCA locations for a shielded NSS. The data locations for the two actuators are at the axial distance from the core stations zero to the centerline of the SSCVAs and at a radius near that of the permanent magnet of each SSCVA.

Table 10-4 presents the uncertainty factors associated with the data presented in Table 10-1.

The graphite core fast ($E > 1.0 \text{ MeV}$) neutron fluxes are slightly greater (3 percent) than those for the composite core. This is the reverse effect observed at the CDDA locations where the composite core flux was greater than that for the graphite. This is attributed to differences in the analytical techniques. The SSCVA data were obtained with point kernel calculations and the CDDA were obtained from transport calculations.

The composite core intermediate neutron fluxes are slightly greater (1 and 3 percent) than those for the graphite core. The converse was true for the fast neutron fluxes and is the same trend observed for the CDDA data. This difference is due to the application of the transport calculated ratio of intermediate to fast fluxes to the point kernel calculated fast flux. The transport calculated fast and intermediate fluxes external to the NSS were greater for the composite than graphite cores causing the composite intermediate fluxes to be greater than those for the graphite core at the SSCVA locations.

The graphite core thermal neutron fluxes are slightly greater (5 percent) than those for the composite core. This is the same trend as that for the fast neutron flux and the reverse of that obtained at the CDDA locations where the composite was slightly higher than the graphite. This difference at the SSCVA and CDDA comparisons is due to the use of a thermal to fast neutron ratio obtained from transport calculations. The difference in this ratio for the composite and graphite cores was close to one percent, thus the thermal fluxes reflect the difference due to the point kernel calculated fast neutron fluxes at the SSCVA locations.

The change in ratio of the composite to graphite neutron fluxes from less than one to greater than one for the three neutron energy groups is attributed to the coupling of point kernel, and transport calculations to obtain the data at the SSCVAs. The differences in the neutron data for the composite and graphite cores is so small that they may be considered to be equal.

The graphite core gamma dose levels are slightly greater (5 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

Additional flux and dose rates for the SSCA can be obtained from the isoflux and isodose rate data presented in Section 2.2.2.3.

The derivation of the uncertainties presented in Table 10-4 is presented in Section 2.4.4 with a discussion of the application of these values to the reliability analysis of the SSCA presented in Section 10.4.4.2. These data are to be used to ensure that the design of the SSCA will attain the 0.99990 probability of success for operation with the internal shield in position.

10.5.1.1.1.2 Fluence and Dose

Table 10-1 presents the steady state neutron fluences and dose levels for one-hour and ten-hour operation in space at 1511 MW. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma data are used to assist in the selection of materials that are capable of reliable performance in these environments. They are also used to determine fluence levels to which materials are to be irradiated in radiation effects tests of materials.

The uncertainty factors associated with the data presented in Table 10-1 are presented in Table 10-4.

Additional fluence and gamma dose data at other locations of the SSCA ECC can be obtained by multiplying the isoflux data in Section 2.2.2.3 by 3.6×10^3 and 3.6×10^4 to obtain one-hour and ten-hour fluence. The isodose data are to be multiplied by 1 and 10 to obtain one- and ten-hour dose levels, respectively. Since the calculated difference between the graphite and composite data is small for the external environment, the data are applicable to graphite and composite cores.

10.5.1.1.2 Radiation Heating Rates

Table 10-1 presents a summary of the nominal heating rate data at the selected SSCA locations. This table provides the following information related to the heating rate:

The heating rate (watts/gm) for stainless steel 304.

To obtain the heating rate in BTU/hr-in³, the watts/gm data are to be multiplied by 443.

Table 10-4 presents the uncertainty factors associated with the heating rates.

The graphite core heating rates are slightly greater (4 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

10.5.1.2 Transient

Table 10-4 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the SSCA selected locations. Table 10-4 contains the following information:

1. The thermal neutron fluence ($E < 0.4 \text{ eV}$)
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$)
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$)
4. The integral radiation heating (watt-sec/gm)
5. The gamma dose (Rad-C).

Table 10-4 also includes the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4.2 discusses the method by which these factors were obtained.

The transient data include the effect of fission product buildup for various operating times. Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.1.2.

10.5.2 Graphite Core

10.5.2.1 Steady State

10.5.2.1.1 Radiation Environment

10.5.2.1.1.1 Fluxes and Dose Rates

Table 10-2 presents the steady state neutron fluxes and dose rates after one-hour operation at 1511 MW thermal power operation in space for several SSCA locations for a shielded NSS. The data locations for the two actuators are at the axial distance from the core station zero to the centerline of the SSCVAs and at a radius near that of the permanent magnet of each SSCVA.

Table 10-5 presents the uncertainty factors associated with the data presented in Table 10-2.

The graphite core fast ($E > 1.0 \text{ MeV}$) neutron fluxes are slightly greater (3 percent) than those for the composite core. This is the reverse effect observed at the CDDA locations where the composite core flux was greater than that for the graphite. This is attributed to differences in the analytical techniques. The SSCVA data were obtained with point kernel calculations and the CDDA were obtained from transport calculations.

The composite core intermediate neutron fluxes are slightly greater (1 and 3 percent) than those for the graphite core. The converse was true for the fast neutron fluxes and is the same trend observed for the CDDA data. This difference is due to the application of the transport calculated ratio of intermediate to fast fluxes to the point kernel calculated fast flux. The transport calculated fast and intermediate fluxes external to the NSS were greater for the composite than graphite cores causing the composite intermediate fluxes to be greater than those for the graphite core at the SSCVA locations.

The graphite core thermal neutron fluxes are slightly greater (5 percent) than those for the composite core. This is the same trend as those for the fast neutron flux and the reverse of that obtained at the CDDA locations where the composite was slightly higher than the graphite. This difference at the SSCVA and CDDA comparisons is due to the use of a thermal to fast neutron ratio obtained from transport calculations.

The difference in this ratio for the composite and graphite cores was close to one percent, thus the thermal fluxes reflect the difference due to the point kernel calculated fast neutron fluxes at the SSCVA locations.

The change in ratio of the composite to graphite neutron fluxes from less than one to greater than one for the three neutron energy groups is attributed to the coupling of point kernel, and transport calculations to obtain the data at the SSCVAs. The difference in the neutron data for the composite and graphite cores is so small that they may be considered to be equal.

The graphite core gamma dose levels are slightly greater (5 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

Additional flux and dose rates for the SSCA can be obtained from the isoflux and isodose rate data presented in Section 2.2.2.3.

The derivation of the uncertainties presented in Table 10-5 is presented in Section 2.4.4 with a discussion of the application of these values to the reliability analysis of the SSCA presented in Section 10.4.4.2. These data are to be used to ensure that the design of the SSCA will attain the 0.99990 probability of success for operation with the internal shield position.

10.5.2.1.1.2 Fluence and Dose

Table 10-2 presents the steady state neutron fluences and dose levels for one-hour and ten-hour operation in space at 1511 MW. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma data are used to assist in the selection of materials that are capable of reliable performance in these environments. They are also used to determine fluence levels to which materials are to be irradiated in radiation effects tests of materials.

The uncertainty factors associated with the data presented in Table 10-2 are presented in Table 10-5.

Additional fluence and gamma dose data at other locations of the SSCA ECC can be obtained by multiplying the isoflux data in Section 2.2.2.3 by 3.6×10^3 and 3.6×10^4 to obtain one-hour and ten-hour fluence. The isodose data are to be multiplied by 1 and 10 to obtain one-and ten-hour dose levels, respectively. Since the calculated difference between the graphite and composite core data is small for the external environment, the data are applicable to graphite and composite cores.

10.5.2.1.2 Radiation Heating Rates

Table 10-2 presents a summary of the nominal heating rate data at the selected SSCA locations. This table provides the following information related to the heating rate:

The heating rate (watts/gm) for stainless steel 304.

To obtain the heating rate in BTU/hr-in³, the watts/gm data are to be multiplied by 443.

Table 10-5 presents the uncertainty factors associated with the heating rates.

The graphite core heating rates are slightly greater (4 percent) than those for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

10.5.2.2 Transient

Table 10-5 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the SSCA selected locations. Table 10-5 contains the following information:

1. The thermal neutron fluence ($E < 0.4$ eV)
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$)
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$)
4. The integral radiation heating (watt - sec/gm)
5. The gamma dose (Rad-C).

Table 10-5 also includes the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4.2 discusses the method by which these factors were obtained.

The transient data include the effect of fission product buildup for various operating times.

Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.2.2.

10.5.3 Unshielded Core

10.5.3.1 Radiation Environment

10.5.3.1.1 Steady State

Table 10-3 presents the radiation environment at the SSCA locations for operation with the internal shield removed. Table 10-6 includes the uncertainty factors for operation in this mode. These data are calculated with the SSCA moved 13 inches closer to the core when the shield is removed. This is the same distance that the dome flange moves when the shield is removed⁽²⁾.

It is observed that the removal of the internal shield plus moving the SSCA 13 inches closer to the core increases the fast, thermal and intermediate neutron fluxes by factors of

84.3, 348, and 160 respectively, at the actuator locations. Similarly, the gamma dose rate is increased by a factor of 16.2. These are obtained as the ratio of the unshielded data to the graphite core data. These ratios are different than those obtained for the CDDA actuator locations. This difference is attributed to the combination of 1) the SSCA locations have a large contribution of radiation coming from the leakage along the radial pressure vessel surface which is not significantly affected by the removal of the shield, 2) the use of point kernel for the SSCVA locations which are located in a region where the point kernel technique is limited, and 3) the intermediate to fast and thermal to fast ratios were not obtained from MAP calculations, but were obtained from one-dimensional axial ANISN calculations which does not consider the radial pressure vessel neutron leakage contribution at the SSCVA. Additional calculations are required to obtain improved data at the SSCVA for an unshielded reactor.

10.5.3.1.2 Transient

Table 10-6 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. These data were obtained by the methods outlined in Section 2.4. These data are provided at the selected SSCA location. Table 10-6 provides the following information:

1. The thermal neutron fluence ($E < 0.4 \text{ eV}$)
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$)
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$)
4. The gamma dose (Rad-C)
5. The integral radiation heating (watt-sec/gm)

This table also presents the uncertainty factors associated with the data.

These transient data include the effect of fission product buildup for various operating times.

10.6 REFERENCES

1. "Structural Support Coolant Valve Actuator, Layout of", Drawing 940J510, September 29, 1970.
2. "SSCA, Internal and Shield Baffle, Study Layout of," Drawing 940J420, September 25, 1970.
3. John Burke, "Structural Support Coolant Valve Actuator Failure Mode Analysis," DRM-52408, September 30, 1970. To be updated.
4. R. Schatz, "NSS Reliability Analysis Model-Preliminary R-202 Report in Support of Engine PDR," DRM-53021, August 6, 1970. To be updated.

TABLE 10-1

NOMINAL RADIATION ENVIRONMENT AT SEVERAL SSCA LOCATIONS
FOR THE COMPOSITE CORE (DWG. 939J723)

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)
Structural Support	49.0	-103.0	6.24(-3)	1.72(6)	1.72(6)	1.72(7)
Coolant Valve	42.5	-111.05	3.53(-3)	9.72(5)	9.72(5)	9.72(6)
Actuators						

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
Structural Support	4.32(10)	1.56(14)	1.56(15)	2.37(11)	8.53(14)	8.53(15)	2.71(10)	9.76(13)	9.76(14)
Coolant Valve	3.40(10)	1.22(14)	1.22(15)	1.88(11)	6.77(14)	6.77(15)	2.79(10)	1.00(14)	1.00(15)
Actuators									

NOTE: Numbers in parentheses refer to powers of ten, i. e., 6.24(-3) $\equiv$ 6.24 $\times$ 10⁻³.

789 10/05/70

239 10/05/70 590 10/06/70

242 10/05/70 591 10/06/70

132 10/02/70

TABLE 10-2

NOMINAL RADIATION ENVIRONMENT AT SEVERAL SSCA LOCATIONS
FOR THE GRAPHITE CORE (DWG. 939J723)

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)
Structural Support Coolant Valve Actuators	49.0	-103.0	6.51(-3)	1.80(6)	1.80(6)	1.80(7)
	42.5	-111.05	3.72(-3)	1.03(6)	1.03(6)	1.03(7)

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
Structural Support Coolant Valve Actuators	4.52(10)	1.63(14)	1.63(15)	2.35(11)	8.46(14)	8.46(15)	2.79(10)	1.00(14)	1.00(15)
	3.57(10)	1.29(14)	1.29(15)	1.84(11)	6.62(14)	6.62(14)	2.88(10)	1.04(14)	1.04(15)

NOTE: Numbers in parentheses refer to powers of ten, i. e., 6.51(-3) = 6.5×10^{-3} .

119 10/02/70 471 10/12/70
240 10/05/70 450 10/12/70
429 10/12/70

TABLE 10-3

NOMINAL RADIATION ENVIRONMENT AT SEVERAL SSCA LOCATIONS
FOR OPERATION IN SPACE WITH THE INTERNAL SHIELD REMOVED
(DWG. 940J420)

Location	Position		Gamma Environment			
	Radius (in.)	Station (in.)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose
			(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)
Structural Support Coolant Valve Actuators	49.0	-90.0	9.52 (-2)	2.92(7)	2.92 (7)	2.92 (8)
	42.5	-98.05	9.29 (-2)	2.86(7)	2.86 (7)	2.86 (8)

Location	Neutron Environment								
	E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
Structural Support Coolant Valve Actuators	(n/cm ² -sec)	(γ/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
	1.57(13) 1.68(13)	5.65 (16) 6.04 (16)	5.65 (17) 6.04 (17)	3.79 (13) 4.05 (13)	1.36(17) 1.46(17)	1.36(18) 1.46(18)	2.35(12) 2.51(12)	8.45 (15) 9.04(15)	8.45(16) 9.04(16)

NOTE: Numbers in parentheses refer to powers of ten, i.e., $9.52(-2) \equiv 9.52 \times 10^{-2}$.

403 11/05/70

429 11/05/70

TABLE 10-4

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL STRUCTURAL SUPPORT COOLANT ASSEMBLY LOCATIONS FOR THE COMPOSITE FUELED CORE

LOCATION - SSCV ACTUATOR-COMPOSITE
RADIUS (IN) 49.00 STATION (IN) -103.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.03E-01	1.40	3.60E 00	1.40	3.62E 00	1.40	3.78E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.47E 11	3.10	1.55E 13	3.10	1.55E 13	3.10	1.62E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.91E 12	3.10	1.35E 14	3.10	1.36E 14	3.10	1.42E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	7.12E 11	3.10	2.46E 13	3.10	2.48E 13	3.10	2.59E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	7.89E 03	1.40	2.75E 05	1.40	2.76E 05	1.40	2.89E 05	1.40

LOCATION - SSCV ACTUATOR-COMPOSITE
RADIUS (IN) 42.50 STATION (IN) -111.05

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	5.85E-02	1.40	2.03E 00	1.40	2.05E 00	1.40	2.14E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.60E 11	3.10	1.59E 13	3.10	1.60E 13	3.10	1.67E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.10E 12	3.10	1.07E 14	3.10	1.08E 14	3.10	1.13E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	5.60E 11	3.10	1.94E 13	3.10	1.95E 13	3.10	2.04E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.46E 03	1.40	1.55E 05	1.40	1.56E 05	1.40	1.63E 05	1.40

TABLE 10-5

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL STRUCTURAL SUPPORT COOLANT ASSEMBLY
LOCATIONS FOR THE GRAPHITE FUELED CORE

LOCATION - SSCV ACTUATOR-GRAPHITE
RADIUS (IN) 49.00 STATION (IN) -103.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.07E-01	1.40	3.74E 00	1.40	3.76E 00	1.40	3.93E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.60E 11	3.10	1.59E 13	3.10	1.60E 13	3.10	1.67E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.22E 12	3.10	7.70E 13	3.10	7.74E 13	3.10	8.09E 13	3.10
NEUTRONS/CM2, E LT 0.4EV	7.45E 11	3.10	2.58E 13	3.10	2.59E 13	3.10	2.71E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	8.25E 03	1.40	2.88E 05	1.40	2.89E 05	1.40	3.02E 05	1.40

LOCATION - SSCV ACTUATOR-GRAPHITE
RADIUS (IN) 42.50 STATION (IN) -111.05

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	6.14E-02	1.40	2.14E 00	1.40	2.15E 00	1.40	2.25E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.75E 11	3.10	1.64E 13	3.10	1.65E 13	3.10	1.73E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.03E 12	3.10	1.05E 14	3.10	1.05E 14	3.10	1.10E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	5.88E 11	3.10	2.04E 13	3.10	2.05E 13	3.10	2.14E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.72E 03	1.40	1.65E 05	1.40	1.65E 05	1.40	1.73E 05	1.40

TABLE 10-6

TIME INTEGRATED RADIATION ENVIRONMENT FOR SEVERAL STRUCTURAL SUPPORT COOLANT ASSEMBLY LOCATIONS FOR AN UNSHIELDED CORE

LOCATION - SSCV-1 ACTUATOR - UNSHIELDED
 RADIUS (IN) 49.00 STATION (IN) -90.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.52E 00	2.00	5.42E 01	2.00	5.45E 01	2.00	5.70E 01	2.00
NEUTRONS/CM2, E GT 1.0MEV	3.87E 13	5.00	1.34E 15	5.00	1.35E 15	5.00	1.41E 15	5.00
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.25E 14	5.00	2.16E 16	5.00	2.17E 16	5.00	2.27E 16	5.00
NEUTRONS/CM2, E LT 0.4EV	2.59E 14	5.00	8.95E 15	5.00	9.00E 15	5.00	9.41E 15	5.00
GAMMA RAY DOSE, RADS (CARBON)	1.30E 05	2.00	4.62E 06	2.00	4.65E 06	2.00	4.86E 06	2.00

LOCATION - SSCV-2 ACTUATOR - UNSHIELDED
 RADIUS (IN) 42.50 STATION (IN) -98.05

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM- STAINLESS STEEL	1.48E 00	2.00	5.29E 01	2.00	5.32E 01	2.00	5.56E 01	2.00
NEUTRONS/CM2, E GT 1.0MEV	4.14E 13	5.00	1.43E 15	5.00	1.44E 15	5.00	1.50E 15	5.00
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.67E 14	5.00	2.31E 16	5.00	2.32E 16	5.00	2.43E 16	5.00
NEUTRONS/CM2, E LT 0.4EV	2.77E 14	5.00	9.58E 15	5.00	9.63E 15	5.00	1.01E 16	5.00
GAMMA RAY DOSE, RADS (CARBON)	1.27E 05	2.00	4.53E 06	2.00	4.55E 06	2.00	4.76E 06	2.00

11.0 NUCLEAR SUBSYSTEM INSTRUMENTATION

11.1 INTRODUCTION

The Nuclear Subsystem Instrumentation (NSSI) shall consist of pressure transducers, thermocouples, displacement transducers, strain transducers, accelerometers, neutronic instruments, and the associated pressure tubing, mounting hardware, and electrical leads to the first termination exterior to the pressure vessel. The components of the NSSI are used to measure selected parameters for the purpose of control, performance diagnostics, safety diagnostics, and collection of trend data.

11.1.1 Description of Nuclear Subsystem Instrumentation

The Nuclear Subsystem Instrumentation (NSSI) consists of thermocouples, transducers for pressure, displacement, strain and acceleration, neutronic instruments, and associated tubing mounting hardware and electrical leads. These instruments are used to measure selected parameters for the purpose of control, performance diagnostics, safety diagnostics and the collection of trend data⁽¹⁾. The NSSI shall sense the parameters throughout all operating conditions including malfunction and emergency modes, specified in CP-677555.

The list and description of the instrumentation for the flight engine is presented in Table 1 of Reference 2 with revisions listed in Reference 3. This instrumentation list⁽²⁾ does not define the locations of the sensors. The instrumentation locations used herein, however, are consistent with those used in Trade Study (T/S) 792⁽⁴⁾ currently in progress. The range of data provided will be used in T/S 792 as a radiation environment requirement for evaluation of transducer selection. The instrumentation locations are preliminary in nature and subject to change based upon the results of T/S 792. The locations were chosen in order to provide a range in the radiation environment which is anticipated to provide the maximum levels.

The instrumentation locations for the two structural support coolant valve (SSCV) position indicators were scaled from ANSC drawing 1137400, Revision C.

11.1.2 Content of Chapter

This chapter provides the nominal (uncertainties not applied) radiation environment for the shielded graphite and composite cores. The NSSI are located at many locations and are not associated with any particular ECC.

The neutron environment data provided for steady state operation of the NSS at 1511 MW thermal power are the neutron flux plus one-and ten-hour fluence for 1) fast neutrons ($E \geq 1.0 \text{ MeV}$), 2) intermediate neutrons ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) and 3) thermal neutrons ($E < 0.4 \text{ eV}$). The photon environment data provided for steady state operation are the photon dose rate (Rad-C/Hr) plus one-and ten-hour dose levels (Rad-C) and the heating rate (watt/gm) for selected materials. The transient data presented in this chapter provide similar time integrated information for the power profile presented in Figure 2-161. The uncertainties in these data are also presented for these nominal data. These data are provided for both the shielded graphite and composite cores. No photon source data will be presented in this chapter since the instrumentation was not considered as a source of secondary photons.

Additional information supplied is related to the following:

1. The requirements on the diagnostic and control instrumentation.
2. A brief description of the analytical model, techniques, and validity of technique employed to calculate the radiation results for the nuclear diagnostic and control instrumentation ECC.
3. The status/maturity of the data and presentation of the results and analysis, including uncertainties associated with the radiation environment and heating rates for this ECC.

11.2 SUMMARY OF RESULTS AND CONCLUSIONS

The radiation environment data presented in this chapter are to be used in Trade Study 792⁽⁴⁾. The data define the radiation environmental requirements of the NSSI.

This section presents the maximum nominal (uncertainty factors not applied) radiation levels to be encountered by any of the NSSI at the locations selected for shielded composite and graphite cores. Data are in Chapters 9.0 and 10.0 at the CDDA and SSCV instrumentation locations (same as those for actuators) for an unshielded NSS

and at the CDDA location for operation in Test Cell C. The radiation data are summarized according to fast neutron ($E > 1.0 \text{ MeV}$), intermediate neutron ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) and thermal neutron ($E < 0.4 \text{ eV}$) environment plus the heating and gamma dose rate data. The flux and dose rate data presented are after one-hour operation at 1511 MW thermal power.

11.2.1 Fast Neutron ($E > 1.0 \text{ MeV}$) Environment

The maximum nominal fast neutron flux for any of the defined instrumentation locations is $4.76 \times 10^{15} (1.03)^*$ and $4.51 \times 10^{15} (1.02) \text{ n/cm}^2\text{-sec}$ for the composite and graphite cores, respectively. The corresponding ten-hour fluences are $1.71 \times 10^{20} (1.03)$ and $1.62 \times 10^{20} (1.02) \text{ n/cm}^2$. These levels are for the core unfueled element and structural support liner temperature sensor locations at a radius of 0.0 inches and station 26.0 inches (the core center). The composite core flux is slightly (6 percent) greater than those for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core.

11.2.2 Intermediate Neutron ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$) Environment

The maximum nominal intermediate neutron flux for any of the defined instrumentation locations is $2.37 \times 10^{16} (1.03)$ and $2.29 \times 10^{16} (1.02) \text{ n/cm}^2\text{-sec}$ for the composite and graphite core, respectively. The corresponding ten-hour fluences are $8.53 \times 10^{20} (1.03)$ and $8.25 \times 10^{20} (1.02) \text{ n/cm}^2$. These levels are for the core unfueled element and structural support liner temperature sensor locations at a radius of 0.0 inches and station 26.0 inches (the core center). The composite core flux is slightly (4 percent) greater than those for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core.

11.2.3 Thermal Neutron ($E < 0.4 \text{ eV}$) Environment

The maximum nominal thermal neutron flux for any of the defined instrumentation locations is $5.11 \times 10^{14} (1.35)$ and $5.45 \times 10^{14} (1.29) \text{ n/cm}^2\text{-sec}$ for the composite and graphite cores, respectively. The corresponding ten-hour fluences are $1.84 \times 10^{19} (1.35)$ and $1.96 \times 10^{19} (1.29) \text{ n/cm}^2$. These levels are for the reflector material temperature sensor location at a radius of 22.2 inches and station of 26.0 inches.

* Number in parenthesis in Section 11.2 denote the one sigma uncertainty factor that apply to the nominal data.

The graphite core flux is slightly greater (7 percent) than the composite core. This is attributed to the more efficient moderating properties of the graphite core.

11.2.4 Gamma Dose Environment

The maximum nominal gamma dose (frequently referred to as Kerma) rate for any of the defined instrumentation locations is 2.46×10^{10} (1.14) and 3.39×10^{10} (1.12) Rad-C/Hr for the composite and graphite cores, respectively. The corresponding ten-hour dose levels are 2.46×10^{11} (1.14) and 3.39×10^{11} (1.12) Rad-C. These levels are for the core unfueled element and structural support liner temperature sensor locations at a radius of 0.0 inches and station of 26.0 inches (the core center). The maximum graphite core dose rate is 38 percent greater than that for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

11.2.5 Radiation Heating Environment

The maximum nominal heating rate environment for any of the defined instrumentation locations is 6.88×10^1 (1.14) and 9.45×10^1 (1.12) watts per gram of stainless steel 304 for the composite and graphite fueled cores, respectively. These levels are for the core unfueled element and structural support liner temperature sensor locations at the core center. The maximum graphite core heating rate is 38 percent greater than that for the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core.

11.3 REQUIREMENTS

This section specifies governing specifications and the applicable requirements in the specification of the NSSI which require radiation information.

11.3.1 Specification Identification

The governing specification for the Nuclear Subsystem Instrumentation ECC is EC-677565.

11.3.2 Applicable Requirements

"3.1.2.1 Reliability. - The reliability allocation for the NSSI is 0.99990.

3.1.2.4.2 Induced Environment.

3.1.2.4.2.1 Nuclear Environment. - The NSSI shall be designed to withstand the induced nuclear environment specified in Tables 2 and 3.

3.1.2.4.2.3 Thermal Environment. - The NSSI shall be designed to withstand the induced thermal environment specified in Table 4 (tbd).

3.3.3 Materials, Parts, and Processes. - The NSSI design shall be in accordance with the requirements of 3.3.3 (Materials, Parts, and Processes) of CP-677555. "

11.4 METHOD OF ANALYSIS - DESIGN PRACTICES

A brief description of the analytical model, analytical procedure, status/maturity of the input data and the validity of the analysis is presented in this section.

11.4.1 Analytical Procedures

The analytical procedure used to generate the radiation environment for the shielded and unshielded graphite and composite cores is presented in Section 2.4.2.

11.4.2 Analytical Models

Since the NSSI is located throughout the entire Nuclear Subsystem, the analytical model employed for the entire NSS analysis is applicable. This analytical model is described in Section 2.4.1. The instrumentation mass was not included in the geometrical model since it is not available and it is expected to be small, creating negligible perturbations in the calculated radiation environment.

11.4.3 Status/Maturity of Input Data

The status/maturity of the primary data required to perform the radiation computations at the instrumentation locations for the graphite and composite cores is presented in Section 2.4.3.

The status/maturity of other requirements is as follows:

The instrumentation list was obtained from Reference 2 with the changes presented in Reference 3 being incorporated. It is anticipated that further adjustments will be made based upon the R/E test results. The changes resulted in the deletion of the following items:

<u>Item No.</u>	<u>Measurement</u>
6	Structural support end cap material temperature
7	Structural support aft end gas temperature
18	Core axial displacement
19	Core radial displacement
22	Core inlet plenum pressure
23	Core exit plenum pressure

2. Measurement of the gamma dose rate at the outside surface of the pressure vessel at the core midplane has been subsequently added to the list.
3. The detailed description of the instrumentation as to size, material composition, wiring description and location was not available. Thus, heating rates and distribution in the NSS for instrument packages for materials that may be included as part of the instrumentation are not presented, nor were the masses included in the detailed geometrical model of the NSS used in the calculations. This is expected however, to have a minor effect on the calculated radiation environment presented herein.
4. The instrument locations used here are consistent with Trade Study 792⁽⁴⁾. The maximum radiation levels determined become an environmental requirement to be used in T/S792 for evaluation of transducer selection.

11.4.4 Validity of the Analysis

11.4.4.1 Comparison with Experimental Results

A discussion of the comparison of the calculated radiation environment with applicable experimental data is presented in Section 2.4.

11.4.4.2 Uncertainty Analysis

Tables 11-2 and 11-5 include the uncertainty factors associated with the radiation environments presented in Table 11-1 and 11-4, respectively. The derivation of these factors is presented in Section 2.4.

The uncertainty in the radiation environment at the instrumentation locations is considered in the design of the instrumentation.

The instrumentation will be used in the Failure Mode Effects and Criticality Analysis of the NSS in that they will be used to detect malfunctions of the system, unsafe operating conditions and incipient failures. The information obtained from these instruments will assist in the determination of the reliability of the system during the R/E series of tests. The failure of a single instrument will not result in a condition that

will impair the success of a test. There will be several of each type of instrument for redundancy such that the 0.99990 probability of success of meeting the performance characteristics of the NSSI will be attained.

11.5 RESULTS, ANALYSIS, AND INTERPRETATION

Data are presented for the composite and graphite cores for steady state and transient operation of the NSS. Data for unshielded radiation environment at the CDDA (Item 38 in Tables 11-1 and 11-2) and SSCV (Item 37 in Tables 11-1 and 11-2) are those presented in Chapters 9.0 and 10.0, respectively. Data at the CDDA location for operation in Test Cell C are presented in Chapter 9.0. Radiation data are presented as fast ($E > 1.0$ MeV), intermediate ($0.4 \text{ eV} \leq E \leq 1.0$ MeV) and thermal ($E < 0.4 \text{ eV}$) neutron flux plus one- and ten-hour neutron fluences. In addition, heating rates (watts/gm) and photon dose rates (Rad-C/Hr) and dose levels for one- and ten-hour operation are presented. The analytical models, analytical procedure and the status/maturity of the input data used to obtain the results presented in this chapter are discussed in Sections 11.4.2, 11.4.1 and 11.4.3, respectively.

11.5.1 Composite Core

11.5.1.1 Steady State

11.5.1.1.1 Radiation Environment

11.5.1.1.1.1 Fluxes and Dose Rates

Table 11-1 presents the nominal (uncertainties not applied) steady state radiation environment after one-hour operation at 1511 MW thermal power operation in space at the NSSI locations where data are to be obtained. These locations were chosen where the radiation environment includes the maximum levels plus additional locations that provide an indication of the range of radiation levels to be encountered. Table 11-2 presents the uncertainty factors associated with the data presented in Table 11-1.

Table 11-3 presents the maximum nominal fast, intermediate and thermal neutron fluxes for any of the defined instrumentation locations with uncertainty factors of 1.03, 1.03 and 1.35, respectively. The maximum fast and intermediate fluxes occur at the core center where the core unfueled element (Item 2 in Table 11-1) and structural support liner (Item 5 in Table 11-1) temperature sensors are located. The maximum thermal flux occurs at the reflector material temperature sensor location.

The composite core fast and intermediate maximum nominal neutron fluxes are respectively 6 and 4 percent greater than for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core. The reverse is true for the maximum nominal thermal flux, where the graphite core flux in the reflector is 7 percent greater than that for the composite core. This is attributed to the more efficient moderating properties of the graphite core.

The maximum nominal gamma dose rate for any of the defined instrumentation locations is provided in Table 11-3. This location is at the core center where the core unfueled elements and structural support liner temperature sensors are located. The graphite core dose rate is 38 percent greater than that for the composite core. This is attributed to the higher density composite core resulting in a higher attenuation of gamma rays by the core.

Additional flux and dose rate data at any other possible instrumentation locations can be obtained from the data presented in Section 2.5.1.

11.5.1.1.1.2 Fluence and Dose

Table 11-1 presents the neutron fluences and dose levels for one-hour and ten-hour operation in space at 1511 MW thermal power. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma data are used to assist in the selection of sensors that are capable of reliable performance in these environments.⁽⁴⁾ They are also used to determine fluence levels to which materials are to be irradiated in radiation effects tests of materials. The uncertainty factors associated with the data presented in Table 11-1 are presented in Table 11-2.

The maximum nominal fluence levels for ten-hour operation for any of the instrumentation locations are 1.71×10^{20} , 8.53×10^{20} and 1.84×10^{19} n/cm² for fast, intermediate and thermal neutrons, respectively. The corresponding uncertainty factors are 1.03, 1.03 and 1.35.

The maximum nominal dose level for ten-hour operation for any instrumentation locations is 2.46×10^{11} Rad-C with a respective uncertainty factor of 1.14.

11.5.1.1.2 Radiation Heating Rates

Table 11-1 presents a summary of the nominal heating rate data at the NSSI locations. Table 11-2 presents the uncertainty factors associated with the data. Table 11-1 provides the heating rate (watts/gm) for stainless steel 304 after one-hour of continuous operation at 1511 MW. This was chosen since it appears as a likely candidate for material selection. After completion of T/S 792 a reevaluation of these data will be made to determine if additional material heating rates are required. To obtain the heating rate in BTU/hr-in³, the watts/gm data are to be multiplied by 443.

Table 11-3 presents the maximum nominal heating rate for any of the instrumentation locations being 68.8 watts/gm (SS) after one-hour of continuous operation of 1511 MW. The uncertainty factor is 1.14. These levels are for the core unfueled element and structural support liner temperature sensor locations at the core center. The maximum graphite core heating rate is 38 percent greater than the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays by the core. Additional heating rate data can be obtained from Section 2.5.1.

11.5.1.2 Transient

Table 11-2 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the selected locations for the NSSI. Table 11-2 contains the following information:

1. The thermal neutron fluence ($E < 0.4$ eV)
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0$ MeV)
3. The fast neutron fluence ($E > 1.0$ MeV)
4. The integral radiation heating (watt-sec/gm)
5. The gamma dose (Rad-C)

Table 11-2 also includes the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4.2 discusses the method by which these factors were obtained.

The transient data include the effect of fission product buildup for various operating times. Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.1.2.

11.5.2 Graphite Core

11.5.2.1 Steady State

11.5.2.1.1 Radiation Environment

11.5.2.1.1.1 Fluxes and Dose Rates

Table 11-4 presents the nominal (uncertainties not applied) steady state radiation environment after one-hour operation at 1511 MW thermal power operation in space at the NSSI locations where data are to be obtained. These locations were chosen where the radiation environment includes the maximum levels plus additional locations that provide an indication of the range of radiation levels to be encountered. Table 11-5 presents the uncertainty factors associated with the data presented in Table 11-4.

Table 11-3 presents the maximum nominal fast, intermediate and thermal neutron fluxes for any of the defined instrumentation locations with uncertainty factors of 1.03, 1.03 and 1.35, respectively. The maximum fast and intermediate fluxes occur at the core center where the core unfueled element (Item 2 in Table 11-4) and structural support liner (Item 5 in Table 11-4) temperature sensors are located. The maximum thermal flux occurs at the reflector material temperature sensor location.

The composite core fast and intermediate maximum nominal neutron fluxes are respectively 6 and 4 percent greater than for the graphite core. This is attributed to the lower carbon to uranium ratio in the composite fuel which results in a higher median energy of fission and harder spectrum in the core than for the graphite core. The reverse is true for the maximum nominal thermal flux, where the graphite core flux in the reflector is 7 percent greater than that for the composite core. This is attributed to the more efficient moderating properties of the graphite core.

The maximum nominal gamma dose rate for any of the defined instrumentation

locations is also provided in Table 11-3. This location is at the core center where the core unfueled element and structural support liner temperature sensors are located. The graphite core dose rate is 38 percent greater than that for the composite core. This is attributed to the higher density composite core resulting in a higher attenuation of gamma rays.

Additional flux and dose rate data at other possible instrumentation locations can be obtained from the data presented in Section 2.5.2.

11.5.2.1.1.2 Fluence and Dose

Table 11-4 presents the neutron fluences and dose levels for one-hour and ten-hour operation in space at 1511 MW thermal power. The one-hour fluence levels are obtained from one-hour operation at rated power. The ten-hour data are obtained from 60 ten minute cycles for the operating profile given in Figure 2-161.

The neutron fluence data and the gamma data are used to assist in the selection of sensors that are capable of reliable performance in these environments.⁽⁴⁾ They are also used to determine fluence levels to which materials are to be irradiated in radiation effects tests.

The uncertainty factors associated with the data presented in Table 11-4 are presented in Table 11-5.

The maximum nominal fluence levels for ten-hour operation for any of the instrumentation locations are 1.62×10^{20} , 8.25×10^{20} and 1.96×10^{19} n/cm² for fast, intermediate and thermal neutrons, respectively. The corresponding uncertainty factors are 1.02, 1.02 and 1.29.

The maximum nominal dose level for ten-hour operation for any instrumentation locations is 3.39×10^{11} Rad-C with a respective uncertainty factor of 1.12.

11.5.2.1.2 Radiation Heating Rates

Table 11-4 presents a summary of the nominal heating rate data at the NSSI locations. Table 11-5 presents the uncertainty factors associated with the data. Table 11-4

provides the heating rate (watts/gm) for stainless steel 304 after one-hour of continuous operation at 1511 MW. This was chosen since it appears as a likely candidate for material selection. After completion of TS/792 a reevaluation of these data will be made to determine if additional material heating rates are required.

To obtain the heating rate in BTU/hr-in³, the watts/gm data are to be multiplied by 443.

Table 11-3 presents the maximum nominal heating rate for any of the instrumentation locations being 94.5 watts/gm (ss) after one-hour of continuous operation of 1511 MW. The uncertainty factor is 1.12. These levels are for the core unfueled element and structural support liner temperature sensor locations at the core center. The maximum graphite core heating rate is 38 percent greater than the composite core. This is attributed to the higher density composite fuel resulting in a higher attenuation of the gamma rays.

Additional heating rate data can be obtained from Section 2.5.2.

11.5.2.2 Transient

Table 11-5 provides the radiation environment information at the transient points on the power profile given by Figure 2-161. The data were obtained by the methods outlined in Section 2.4. These data are provided at the selected locations for the NSSI. Table 11-5 contains the following information:

1. The thermal neutron fluence ($E < 0.4 \text{ eV}$)
2. The intermediate neutron fluence ($0.4 \text{ eV} \leq E \leq 1.0 \text{ MeV}$)
3. The fast neutron fluence ($E > 1.0 \text{ MeV}$)
4. The integral radiation heating rates (watt-sec/gm)
5. The gamma dose (Rad-C)

Table 11-5 also includes the one sigma standard deviation uncertainty factors for the radiation data. Section 2.4.4 discusses the method by which these factors were obtained.

The transient data include the effect of fission product buildup for various operating times.

Additional transient data for cyclic operation can be obtained using the technique and integrating factors presented in Section 2.5.2.2.

11.6 REFERENCES

1. A. J. Glaser, "NSS Preliminary Listing of Trend Characteristics Preliminary R-202 Input in Support of Engine PDR," DRM 53024, (August 26, 1970).
2. R. L. Oelrich, "Nuclear Subsystem Instrumentation Measurement Requirement Analysis," DRM 51855A, (August 4, 1970).
3. SI:AEP:1345, A. C. Sanderson to J. L. Dooling; Subject: IED:517A - Flight Engine MRL Input, dated September 18, 1970.
4. WANL-TME-2767, "NSS Instrumentation," Trade Study No. 792, To be published.

TABLE 11-1

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
		Radius (in)	Station (in)	Heating Rate (w/gm(SS))	Dose Rate (Rad-C/hr)	1 Hour Dose (Rad-C)	10 Hour Dose (Rad-C)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)
1	Core Station 0.2 Fuel Element Temp.	0.0	0.2	2.30(1)	8.02(9)	8.02(9)	8.02(10)	1.77(13)	6.37(16)	6.37(17)	6.46(15)	2.33(19)	2.33(20)	1.24(15)	4.46(18)	4.46(19)
		15.75	0.2	1.57(1)	5.50(9)	5.50(9)	5.50(10)	1.21(13)	4.36(16)	4.36(17)	3.88(15)	1.40(19)	1.40(20)	0.81(14)	3.17(18)	3.17(19)
2	Core Unfueled Element Temperature	0.0	18.0	6.67(1)	2.38(10)	2.38(10)	2.38(11)	3.55(12)	1.28(16)	1.28(17)	2.31(16)	8.32(19)	8.32(20)	4.66(15)	1.68(19)	1.68(20)
		0.0	26.0	6.88(1)	2.46(10)	2.46(10)	2.46(11)	3.68(12)	1.32(16)	1.32(17)	2.37(16)	8.53(19)	8.53(20)	4.76(15)	1.71(19)	1.71(20)
		15.75	18.0	4.89(1)	1.75(10)	1.75(10)	1.75(11)	5.92(12)	2.13(16)	2.13(17)	1.40(16)	5.04(19)	5.04(20)	3.39(15)	1.22(19)	1.22(20)
		15.75	26.0	5.05(1)	1.81(10)	1.81(10)	1.81(11)	6.15(12)	2.21(16)	2.21(17)	1.44(16)	5.18(19)	5.18(20)	3.47(15)	1.25(19)	1.25(20)
3	Cluster Exit Gas Temperature	0.0	55.0	3.85(0)	1.33(9)	1.33(9)	1.33(10)	4.19(12)	1.51(16)	1.51(17)	1.15(15)	4.14(18)	4.14(19)	3.35(14)	1.21(18)	1.21(19)
		15.75	55.0	2.57(0)	0.87(8)	0.87(8)	0.87(9)	1.12(13)	4.03(16)	4.03(17)	7.79(14)	2.80(18)	2.80(19)	2.17(14)	7.81(17)	7.81(18)
4	Structural Support Supply Plenum Gas Temperature	0.0	-5.2	9.74(3)	3.20(9)	3.20(9)	3.20(10)	2.37(14)	8.53(17)	8.53(18)	3.50(15)	1.26(19)	1.26(20)	4.52(14)	1.63(18)	1.63(19)
		17.0	-5.2	5.37(0)	1.75(9)	1.75(9)	1.75(10)	1.48(14)	5.33(17)	5.33(18)	1.86(15)	6.70(18)	6.70(19)	2.44(14)	8.78(17)	8.78(18)
5	Structural Support Liner Temperature	0.0	8.0	4.70(1)	1.68(10)	1.68(10)	1.68(11)	2.71(12)	9.76(15)	9.76(16)	1.62(16)	5.83(19)	5.83(20)	3.36(15)	1.21(19)	1.21(20)
		0.0	26.0	6.88(1)	2.46(10)	2.46(10)	2.46(11)	3.68(12)	1.32(16)	1.32(17)	2.37(16)	8.53(19)	8.53(20)	4.76(15)	1.71(19)	1.71(20)
		0.0	50.0	1.63(1)	5.79(9)	5.79(9)	5.79(10)	9.49(11)	3.42(15)	3.42(16)	5.29(15)	1.90(19)	1.90(20)	1.13(15)	4.07(18)	4.07(19)
		16.0	8.0	3.40(1)	1.21(10)	1.21(10)	1.21(11)	5.08(12)	1.83(16)	1.83(17)	9.68(15)	3.48(19)	3.48(20)	2.41(15)	8.68(18)	8.68(19)
		16.0	26.0	4.98(1)	1.78(10)	1.78(10)	1.78(11)	7.53(12)	2.71(16)	2.71(17)	1.41(16)	5.08(19)	5.08(20)	3.40(15)	1.22(19)	1.22(20)
		16.0	50.0	1.17(1)	4.15(9)	4.15(9)	4.15(10)	1.87(12)	6.73(15)	6.73(16)	3.16(15)	1.14(19)	1.14(20)	7.98(14)	2.87(18)	2.87(19)
6	Deleted**															
7	Deleted**															
8	Structural Support Exhaust Plenum Temperature	21.5	-6.5	2.78(0)	8.89(8)	8.89(8)	8.89(9)	1.17(14)	4.21(17)	4.21(18)	8.06(14)	2.90(18)	2.90(19)	9.52(13)	3.43(17)	3.43(18)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3

NOTE: 1) Numbers in parentheses () refer to powers of ten, i. e., 8.91(-3) = 8.91 x 10⁻³
2) Data presented for 1511 MW thermal power.

577	10/08/70	789	10/05/70
441	10/12/70	590	10/06/70
572	10/08/70	591	10/16/70

TABLE 11-1 (Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
9	NSS Inlet from Turbine Gas Temperature	3.0	-32.5	1.87(-2)	5.95(6)	5.95(6)	5.95(7)	9.23(11)	3.32(15)	3.32(16)	4.18(12)	1.50(16)	1.50(17)	9.68(11)	3.48(15)	3.48(16)
10	Reflector Inlet Plenum Temperature	24.5	55.0	1.32(0)	4.44(8)	4.44(8)	4.44(9)	6.93(13)	2.49(17)	2.49(18)	3.57(14)	1.29(18)	1.29(19)	5.14(13)	1.85(17)	1.85(18)
11	NSS Outlet to Turbine Gas Temperature	7.3	-35.0	1.60(-2)	5.09(6)	5.09(6)	5.09(7)	6.52(11)	2.35(15)	2.35(16)	3.11(12)	1.12(16)	1.12(17)	7.91(11)	2.85(15)	2.85(16)
12	Reflector Material Temperature	26.5	4.0	2.76(0)	9.17(8)	9.17(8)	9.17(9)	1.36(14)	4.90(17)	4.90(18)	7.18(14)	2.58(18)	2.58(19)	6.04(13)	2.17(17)	2.17(18)
		26.5	26.0	5.05(0)	1.69(9)	1.69(9)	1.69(10)	2.8(14)	1.01(18)	1.01(19)	1.47(15)	5.29(18)	5.29(19)	1.18(14)	4.25(17)	4.25(18)
		26.5	48.0	1.87(0)	6.25(8)	6.25(8)	6.25(9)	9.08(13)	3.27(17)	3.27(18)	4.83(14)	1.74(18)	1.74(19)	4.11(13)	1.48(17)	1.48(18)
		22.2	4.0	6.29(0)	2.15(9)	2.15(9)	2.15(10)	2.54(14)	9.14(17)	9.14(18)	3.80(15)	1.37(19)	1.37(20)	3.64(14)	1.31(18)	1.31(19)
		22.2	26.0	1.19(1)	4.11(9)	4.11(9)	4.11(10)	5.11(14)	1.84(18)	1.84(19)	7.89(15)	2.84(19)	2.84(20)	7.14(14)	2.57(18)	2.57(19)
		22.2	48.0	4.18(0)	1.44(9)	1.44(9)	1.44(10)	1.65(14)	5.94(17)	5.94(18)	2.52(15)	9.07(18)	9.07(19)	2.42(14)	8.71(17)	8.71(18)
13	Final Seal Exit Gas Temperature	19.5	49.0	6.68(0)	2.35(9)	2.35(9)	2.35(10)	9.34(13)	3.36(17)	3.36(18)	2.84(15)	1.02(19)	1.02(20)	4.52(14)	1.63(18)	1.63(19)
14	Central Shield Material Temperature	0.0	-11.83	3.51(0)	1.16(9)	1.16(9)	1.16(10)	3.92(13)	1.41(17)	1.41(18)	1.00(15)	3.60(18)	3.60(19)	1.20(14)	4.32(17)	4.32(18)
		0.0	-17.5	7.33(-1)	2.43(8)	2.43(8)	2.43(9)	1.27(11)	4.57(14)	4.57(15)	8.44(13)	3.04(17)	3.04(18)	1.77(13)	6.37(16)	6.37(17)
		0.0	-22.93	1.46(-1)	4.68(7)	4.68(7)	4.68(8)	1.95(11)	7.02(14)	7.02(15)	1.29(13)	4.64(16)	4.64(17)	3.34(12)	1.20(16)	1.20(17)
		0.0	-24.23	2.75(-2)	8.68(6)	8.68(6)	8.68(7)	7.36(11)	2.65(15)	2.65(16)	8.72(12)	3.14(16)	3.14(17)	2.19(12)	7.88(15)	7.88(16)
		10.0	-11.83	2.91(0)	9.61(8)	9.61(8)	9.61(9)	3.25(13)	1.17(17)	1.17(18)	8.24(14)	2.97(18)	2.97(19)	9.98(13)	3.59(17)	3.59(18)
		10.0	-17.5	6.12(-1)	2.03(8)	2.03(8)	2.03(9)	1.05(11)	3.78(14)	3.78(15)	6.70(13)	2.41(17)	2.41(18)	1.48(13)	5.33(16)	5.33(17)
		10.0	-22.93	1.23(-1)	3.94(7)	3.94(7)	3.94(8)	1.78(11)	6.41(14)	6.41(15)	1.09(13)	3.92(16)	3.92(17)	2.89(12)	1.04(16)	1.04(17)
		10.0	-24.23	2.37(-2)	7.47(6)	7.47(6)	7.47(7)	6.73(11)	2.42(15)	2.42(16)	7.55(12)	2.72(16)	2.72(17)	1.91(12)	6.88(15)	6.88(16)
		19.4	-11.83	1.52(0)	4.99(8)	4.99(8)	4.99(9)	2.03(13)	7.31(16)	7.31(17)	4.13(14)	1.49(18)	1.49(19)	5.23(13)	1.88(17)	1.88(18)
		19.4	-17.5	3.39(-1)	1.11(8)	1.11(8)	1.11(9)	1.44(11)	5.18(14)	5.18(15)	4.70(13)	1.69(17)	1.69(18)	8.72(12)	3.14(16)	3.14(17)
		19.4	-22.93	7.42(-2)	2.36(7)	2.36(7)	2.36(8)	1.92(11)	6.91(14)	6.91(15)	8.26(12)	2.97(16)	2.97(17)	1.78(12)	6.41(15)	6.41(16)
		19.4	-24.23	2.34(-2)	7.33(6)	7.33(6)	7.33(7)	4.84(11)	1.74(15)	1.74(16)	5.97(12)	2.15(16)	2.15(17)	1.23(12)	4.43(15)	4.43(16)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91(-3) = 8.91 x 10⁻³
2) Data presented for 1511 MW thermal power.

TABLE 11-1 (Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
15	Core Inlet Plenum Temperature	0.0 19.0	-3.0 -3.0	1.48(1) 6.22(0)	4.95(9) 2.06(9)	4.95(9) 2.06(9)	4.95(10) 2.06(10)	1.01(14) 1.07(14)	3.64(17) 3.85(17)	3.64(18) 3.85(18)	4.51(15) 2.01(15)	1.62(19) 7.24(18)	1.62(20) 7.24(19)	7.01(14) 3.00(14)	2.52(18) 1.08(18)	2.52(19) 1.08(19)
16	Support Plate Material Temperature	0.0 8.0 19.0	-6.5 -6.5 -6.5	7.96(0) 7.09(0) 3.71(0)	2.59(9) 2.31(9) 1.19(9)	2.59(9) 2.31(9) 1.19(9)	2.59(10) 2.31(10) 1.19(10)	2.54(14) 2.25(14) 1.38(14)	9.14(17) 8.10(17) 4.97(17)	9.14(18) 8.10(18) 4.97(18)	2.88(15) 2.56(15) 1.22(15)	1.04(19) 9.22(18) 4.39(18)	1.04(20) 9.22(19) 4.39(19)	3.49(14) 3.15(14) 1.48(14)	1.26(18) 1.13(18) 5.33(17)	1.26(19) 1.13(19) 5.33(18)
17	Nuclear Subsystem Vibration	0.0 22.0	-9.5 -7.0	4.94(0) 2.42(0)	1.58(9) 7.71(8)	1.58(9) 7.71(8)	1.58(10) 7.71(9)	1.59(14) 1.07(14)	5.72(17) 3.85(17)	5.72(18) 3.85(18)	1.58(15) 6.65(14)	5.69(18) 2.39(18)	5.69(19) 2.39(19)	1.78(14) 7.62(13)	6.41(17) 2.74(17)	6.41(18) 2.74(18)
18	Deleted**															
19	Deleted**															
20	Structural Support System Load	0.0 15.75	-8.0 -8.0	6.37(0) 3.91(0)	2.06(9) 1.26(9)	2.06(9) 1.26(9)	2.06(10) 1.26(10)	2.29(14) 1.44(14)	8.24(17) 5.18(17)	8.24(18) 5.18(18)	2.22(15) 1.30(15)	7.99(18) 4.68(18)	7.99(19) 4.68(19)	2.55(14) 1.52(14)	9.18(17) 5.47(17)	9.18(18) 5.47(18)
21	Reflector Inlet Plenum Pressure	7.0 19.5	-30.8 -28.8	1.95(-2) 1.45(-2)	6.19(6) 4.56(6)	6.19(6) 4.56(6)	6.19(7) 4.56(7)	8.97(11) 6.62(11)	3.23(15) 2.38(15)	3.23(16) 2.38(16)	4.85(12) 3.90(12)	1.75(16) 1.40(16)	1.75(17) 1.40(17)	1.12(12) 8.15(11)	4.03(15) 2.93(15)	4.03(16) 2.93(16)
22	Deleted**															
23	Deleted**															
24	NSS Outlet to Turbine Pressure	7.0 19.5	-30.8 -28.8	1.95(-2) 1.45(-2)	6.19(6) 4.56(6)	6.19(6) 4.56(6)	6.19(7) 4.56(7)	8.97(11) 6.62(11)	3.23(15) 2.38(15)	3.23(16) 2.38(16)	4.85(12) 3.90(12)	1.75(16) 1.40(16)	1.75(17) 1.40(17)	1.12(12) 8.15(11)	4.03(15) 2.93(15)	4.03(16) 2.93(16)
25	Structural Support Supply Plenum Pressure	7.0 19.5	-30.8 -28.8	1.95(-2) 1.45(-2)	6.19(6) 4.56(6)	6.19(6) 4.56(6)	6.19(7) 4.56(7)	8.97(11) 6.62(11)	3.23(15) 2.38(15)	3.23(16) 2.38(16)	4.85(12) 3.90(12)	1.75(16) 1.40(16)	1.75(17) 1.40(17)	1.12(12) 8.15(11)	4.03(15) 2.93(15)	4.03(16) 2.93(16)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3.

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91(-3) = 8.91 x 10⁻³
2) Data presented for 1511 MW thermal power.

TABLE 11-1 (Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour ² Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
26	NSS Inlet from Turbine Pressure	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)
27	Reflector Inlet/ Turbine Inlet Axial Differential Pressure	7.0 19.5	-30.8 -28.8	1.95(-2) 1.45(-2)	6.19(6) 4.56(6)	6.19(6) 4.56(6)	6.19(7) 4.56(7)	8.97(11) 6.62(11)	3.23(15) 2.38(15)	3.23(16) 2.38(16)	4.85(12) 3.90(12)	1.75(16) 1.40(16)	1.75(17) 1.40(17)	1.12(12) 8.15(11)	4.03(15) 2.93(15)	4.03(16) 2.93(16)
28	Reflector/Inlet Core Outlet Differential Pressure	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)
29	Core Axial Differential Pressure	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)
30	NSS Inlet from Turbine to Core Inlet Differential Pressure	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)
31	Structural Support System Differential Pressure (SSCA Inlet to Reflector Outlet)	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3

NOTE: 1) Numbers in parentheses () refer to powers of ten, i. e., 8.91(-3) = 8.91 x 10⁻³
 2) Data presented for 1511 MW thermal power.

TABLE 11-1 (Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E < 0.4 eV			0.4 eV < E < 1.0 MeV			E > 1.0 MeV		
		Radius (in)	Station (in)	Heating Rate (w/gm(SS))	Dose Rate (Rad-C)	1 Hour Dose (Rad-C)	10 Hour Dose (Rad-C)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)	Flux (n/cm ² -sec)	1 Hour Fluence (n/cm ²)	10 Hour Fluence (n/cm ²)
32	Seal Chamber to Core Exit Differ- ential Pressure	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)
33	Core Nuclear Power (Low Range)	28.08 23.0	26.0 - 4.0	2.86(0) 3.05(0)	9.58(8) 9.87(8)	9.58(8) 9.87(8)	9.58(9) 9.87(9)	8.67(13) 1.33(14)	3.12(17) 4.79(17)	3.12(18) 4.79(18)	4.49(14) 8.99(14)	1.62(18) 3.24(18)	1.62(19) 3.24(19)	5.43(13) 1.06(14)	1.95(17) 3.82(17)	1.95(18) 3.82(18)
34	Core Nuclear Power (Intermediate Range)	28.08 23.0	26.0 - 4.0	2.86(0) 3.05(0)	9.58(8) 9.87(8)	9.58(8) 9.87(8)	9.58(9) 9.87(9)	8.67(13) 1.33(14)	3.12(17) 4.79(17)	3.12(18) 4.79(18)	4.49(14) 8.99(14)	1.62(18) 3.24(18)	1.62(19) 3.24(19)	5.43(13) 1.06(14)	1.95(17) 3.82(17)	1.95(18) 3.82(18)
35	Core Nuclear Power (High Range)	28.08 23.0	26.0 - 4.0	2.86(0) 3.05(0)	9.58(8) 9.87(8)	9.58(8) 9.87(8)	9.58(9) 9.87(9)	8.67(13) 1.33(14)	3.12(17) 4.79(17)	3.12(18) 4.79(18)	4.49(14) 8.99(14)	1.62(18) 3.24(18)	1.62(19) 3.24(19)	5.43(13) 1.06(14)	1.95(17) 3.82(17)	1.95(18) 3.82(18)
36	Gamma Dose Rate	0.0 19.4 28.08	-24.5 -24.5 26.0	2.72(-2) 2.25(-2) 2.86(0)	8.57(6) 7.07(6) 9.58(8)	8.57(6) 7.07(6) 9.58(8)	8.57(7) 7.07(7) 9.58(9)	7.70(11) 5.06(11) 8.67(13)	2.77(15) 1.82(15) 3.12(17)	2.77(16) 1.82(16) 3.12(18)	8.57(12) 5.78(12) 4.49(14)	3.09(16) 2.08(16) 1.62(18)	3.09(17) 2.08(17) 1.62(19)	2.14(12) 1.19(12) 5.43(13)	7.70(15) 4.28(15) 1.95(17)	7.70(16) 4.28(16) 1.95(18)
37	SSCV Position	49.0 42.5	-103.0 -111.05	6.24(-3) 3.53(-3)	1.72(6) 9.72(5)	1.72(6) 9.72(5)	1.72(7) 9.72(6)	4.32(10) 3.40(10)	1.56(14) 1.22(14)	1.56(15) 1.22(15)	2.37(11) 1.88(11)	8.53(14) 6.77(14)	8.53(15) 6.77(15)	2.71(10) 2.79(10)	9.76(13) 1.00(14)	9.76(14) 1.00(15)
38	Control Drum Actuator Shaft Position	24.5	-37.0	8.91(-3)	2.78(6)	2.78(6)	2.78(7)	1.13(11)	4.07(14)	4.07(15)	6.49(11)	2.34(15)	2.34(16)	2.96(11)	1.07(15)	1.07(16)
39***	Fast Neutron Flux	0.0 19.4	-24.5 -24.5	2.72(-2) 2.25(-2)	8.57(6) 7.07(6)	8.57(6) 7.07(6)	8.57(7) 7.07(7)	7.70(11) 5.06(11)	2.77(15) 1.82(15)	2.77(16) 1.82(16)	8.57(12) 5.78(12)	3.09(16) 2.08(16)	3.09(17) 2.08(17)	2.14(12) 1.19(12)	7.70(15) 4.28(15)	7.70(16) 4.28(16)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3

***Additional Measurement

NOTE: 1) Numbers in Parentheses () refer to powers of ten, i. e., 8.91(-3) = 8.91 x 10⁻³.
2) Data presented for 1511 MW thermal power.

TIME INTEGRATED RADIATION ENVIRONMENT AND UNCERTAINTY FACTORS
AT INSTRUMENTATION LOCATIONS FOR THE COMPOSITE FUEL CORE (DWG. 939J723)

LOCATION - CONTROL DRUM ACTUATOR SHAFT POSITION
RADIUS (IN) 24.50 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.48E-01	1.40	5.14E 00	1.40	5.16E 00	1.40	5.39E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.88E 12	3.10	1.69E 14	3.10	1.70E 14	3.10	1.77E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.07E 13	3.10	3.70E 14	3.10	3.72E 14	3.10	3.89E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	1.87E 12	3.10	6.46E 13	3.10	6.49E 13	3.10	6.78E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.28E 04	1.40	4.45E 05	1.40	4.47E 05	1.40	4.67E 05	1.40

LOCATION - STRUCTURAL SUPPORT SYSTEM LOAD
RADIUS (IN) 0.0 STATION (IN) -8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.03E 02	1.09	3.65E 03	1.09	3.67E 03	1.09	3.83E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	4.21E 15	1.71	1.46E 17	1.71	1.46E 17	1.71	1.53E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.67E 16	1.71	1.27E 18	1.71	1.27E 18	1.71	1.33E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	3.77E 15	1.88	1.30E 17	1.88	1.31E 17	1.88	1.37E 17	1.88
GAMMA RAY DOSE, RADS (CARBON)	9.20E 06	1.09	3.26E 08	1.09	3.28E 08	1.09	3.43E 08	1.09

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TABLE 11-2 (Continued)

LOCATION - NUCLEAR SUBSYSTEM VIBRATION
 RADIUS (IN) 0.0 STATION (IN) -9.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.09E 01	1.09	2.84E 03	1.09	2.85E 03	1.09	2.98E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.94E 15	1.71	1.02E 17	1.71	1.02E 17	1.71	1.07E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.60E 16	1.71	8.99E 17	1.71	9.03E 17	1.71	9.44E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	2.63E 15	1.88	9.09E 16	1.88	9.14E 16	1.88	9.55E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	7.17E 06	1.09	2.52E 08	1.09	2.54E 08	1.09	2.65E 08	1.09

LOCATION - STRUCTURAL SUPPORT SYSTEM LOAD
 RADIUS (IN) 15.75 STATION (IN) -8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.35E 01	1.09	2.24E 03	1.09	2.25E 03	1.09	2.35E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.51E 15	1.71	8.68E 16	1.71	8.73E 16	1.71	9.12E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.15E 16	1.71	7.44E 17	1.71	7.48E 17	1.71	7.81E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	2.37E 15	1.88	8.21E 16	1.88	8.26E 16	1.88	8.63E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.63E 06	1.09	1.99E 08	1.09	2.00E 08	1.09	2.10E 08	1.09

TABLE 11-2 (Continued)

LOCATION - NUCLEAR SUBSYSTEM VIBRATION
 RADIUS (IN) 22.00 STATION (IN) -7.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.94E 01	1.09	1.39E 03	1.09	1.39E 03	1.09	1.46E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.26E 15	1.71	4.35E 16	1.71	4.37E 16	1.71	4.57E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.10E 16	1.71	3.79E 17	1.71	3.81E 17	1.71	3.98E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	1.76E 15	1.88	6.09E 16	1.88	6.12E 16	1.88	6.40E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.46E 06	1.09	1.22E 08	1.09	1.23E 08	1.09	1.29E 08	1.09

LOCATION - CORE NUCLEAR POWER(LOW RANGE)
 RADIUS (IN) 28.08 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.27E 01	1.40	1.60E 03	1.40	1.61E 03	1.40	1.68E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.94E 14	3.10	3.09E 16	3.10	3.11E 16	3.10	3.25E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.40E 15	3.10	2.56E 17	3.10	2.57E 17	3.10	2.69E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.94E 16	3.10	4.97E 16	3.10	5.19E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.93E 06	1.40	1.48E 08	1.40	1.49E 08	1.40	1.56E 08	1.40

TABLE 11-2 (Continued)

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

LOCATION - CORE NULLEAR POWER(LOW RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.87E 01	1.14	1.74E 03	1.14	1.75E 03	1.14	1.83E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.75E 15	1.16	6.05E 16	1.16	6.08E 16	1.16	6.36E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.48E 16	1.16	5.13E 17	1.16	5.15E 17	1.16	5.39E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.20E 15	1.35	7.60E 16	1.35	7.63E 16	1.35	7.98E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	4.33E 06	1.14	1.56E 08	1.14	1.57E 08	1.14	1.64E 08	1.14

TABLE 11-2 (Continued)

LOCATION - CLUSTER EXIT GAS TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.66E 01	1.40	2.14E 03	1.40	2.16E 03	1.40	2.26E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	5.51E 15	3.10	1.91E 17	3.10	1.92E 17	3.10	2.00E 17	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.90E 16	3.10	6.57E 17	3.10	6.60E 17	3.10	6.90E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	6.91E 13	3.10	2.39E 15	3.10	2.40E 15	3.10	2.51E 15	3.10
GAMMA RAY DOSE, RADS (CARBON)	5.37E 06	1.40	2.04E 08	1.40	2.05E 08	1.40	2.15E 08	1.40

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.78E 01	1.09	2.02E 03	1.09	2.03E 03	1.09	2.12E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.97E 15	1.71	6.82E 16	1.71	6.86E 16	1.71	7.17E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.65E 16	1.71	5.71E 17	1.71	5.74E 17	1.71	6.00E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	6.47E 14	1.88	2.24E 16	1.88	2.25E 16	1.88	2.35E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.28E 06	1.09	1.85E 08	1.09	1.86E 08	1.09	1.94E 08	1.09

LOCATION - CLUSTER EXIT GAS TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.80E 01	1.40	1.43E 03	1.40	1.44E 03	1.40	1.51E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	3.57E 15	3.10	1.24E 17	3.10	1.24E 17	3.10	1.30E 17	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.28E 16	3.10	4.44E 17	3.10	4.47E 17	3.10	4.67E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.84E 14	3.10	6.37E 15	3.10	6.40E 15	3.10	6.69E 15	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.57E 06	1.40	1.36E 08	1.40	1.36E 08	1.40	1.43E 08	1.40

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -22.93

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.42E 00	1.09	8.40E 01	1.09	8.44E 01	1.09	8.81E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.50E 13	1.71	1.90E 15	1.71	1.91E 15	1.71	2.00E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.13E 14	1.71	7.38E 15	1.71	7.41E 15	1.71	7.75E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	3.22E 12	1.88	1.11E 14	1.88	1.12E 14	1.88	1.17E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.15E 05	1.09	7.49E 06	1.09	7.53E 06	1.09	7.87E 06	1.09

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -17.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.21E 01	1.09	4.22E 02	1.09	4.24E 02	1.09	4.43E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.92E 14	1.71	1.01E 16	1.71	1.02E 16	1.71	1.06E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.39E 15	1.71	4.82E 16	1.71	4.84E 16	1.71	5.06E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	2.10E 12	1.88	7.26E 13	1.88	7.30E 13	1.88	7.63E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.11E 06	1.09	3.89E 07	1.09	3.91E 07	1.09	4.08E 07	1.09

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.80E 01	1.09	1.68E 03	1.09	1.68E 03	1.09	1.76E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.65E 15	1.71	5.69E 16	1.71	5.72E 16	1.71	5.98E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 16	1.71	4.70E 17	1.71	4.72E 17	1.71	4.94E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	5.36E 14	1.88	1.85E 16	1.88	1.86E 16	1.88	1.95E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	4.38E 06	1.09	1.53E 08	1.09	1.54E 08	1.09	1.61E 08	1.09

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -24.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.58E-01	1.09	1.59E 01	1.09	1.60E 01	1.09	1.67E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.60E 13	1.71	1.25E 15	1.71	1.25E 15	1.71	1.31E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.44E 14	1.71	4.97E 15	1.71	5.00E 15	1.71	5.23E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.21E 13	1.88	4.20E 14	1.88	4.22E 14	1.88	4.41E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.99E 04	1.09	1.39E 06	1.09	1.40E 06	1.09	1.46E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -17.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.01E 01	1.09	3.52E 02	1.09	3.54E 02	1.09	3.70E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.44E 14	1.71	8.45E 15	1.71	8.50E 15	1.71	8.88E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.15E 15	1.71	3.99E 16	1.71	4.01E 16	1.71	4.19E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	1.73E 12	1.88	6.00E 13	1.88	6.03E 13	1.88	6.30E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	9.29E 05	1.09	3.24E 07	1.09	3.26E 07	1.09	3.41E 07	1.09

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -22.93

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.03E 00	1.09	7.07E 01	1.09	7.10E 01	1.09	7.42E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	4.76E 13	1.71	1.65E 15	1.71	1.65E 15	1.71	1.73E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.80E 14	1.71	6.24E 15	1.71	6.27E 15	1.71	6.55E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	2.94E 12	1.88	1.02E 14	1.88	1.02E 14	1.88	1.07E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.81E 05	1.09	6.31E 06	1.09	6.34E 06	1.09	6.62E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.51E 01	1.09	8.77E 02	1.09	8.81E 02	1.09	9.21E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	8.61E 14	1.71	2.98E 16	1.71	3.00E 16	1.71	3.13E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.81E 15	1.71	2.36E 17	1.71	2.37E 17	1.71	2.48E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	3.35E 14	1.88	1.16E 16	1.88	1.16E 16	1.88	1.22E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.28E 06	1.09	7.97E 07	1.09	8.01E 07	1.09	8.37E 07	1.09

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -24.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.93E-01	1.09	1.36E 01	1.09	1.37E 01	1.09	1.43E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.14E 13	1.71	1.09E 15	1.71	1.09E 15	1.71	1.14E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.24E 14	1.71	4.31E 15	1.71	4.33E 15	1.71	4.52E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.11E 13	1.88	3.84E 14	1.88	3.86E 14	1.88	4.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.43E 04	1.09	1.19E 06	1.09	1.20E 06	1.09	1.25E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -22.93

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.23E 00	1.09	4.28E 01	1.09	4.30E 01	1.09	4.49E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.94E 13	1.71	1.02E 15	1.71	1.02E 15	1.71	1.07E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 14	1.71	4.71E 15	1.71	4.73E 15	1.71	4.95E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	3.17E 12	1.88	1.10E 14	1.88	1.10E 14	1.88	1.15E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.09E 05	1.09	3.78E 06	1.09	3.80E 06	1.09	3.97E 06	1.09

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -17.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.61E 00	1.09	1.95E 02	1.09	1.96E 02	1.09	2.05E 02	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.44E 14	1.71	4.97E 15	1.71	5.00E 15	1.71	5.22E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.75E 14	1.71	2.68E 16	1.71	2.69E 16	1.71	2.82E 16	1.71
NEUTRONS/CM2, E LT 0.4EV	2.38E 12	1.88	8.72E 13	1.88	8.26E 13	1.88	9.64E 13	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.11E 05	1.09	1.78E 07	1.09	1.79E 07	1.09	1.87E 07	1.09

TABLE 11-2 (Continued)

LOCATION - CORE STATION 0.2 FUEL ELEMENT TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 0.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.37E 02	1.14	1.28E 04	1.14	1.28E 04	1.14	1.35E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	2.04E 16	1.03	7.06E 17	1.03	7.10E 17	1.03	7.42E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.07E 17	1.03	3.69E 18	1.03	3.70E 18	1.03	3.87E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	2.92E 14	1.35	1.01E 16	1.35	1.01E 16	1.35	1.06E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.24E 07	1.14	1.23E 09	1.14	1.24E 09	1.14	1.30E 09	1.14

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -24.73

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.89E-01	1.09	1.35E 01	1.09	1.35E 01	1.09	1.42E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.02E 13	1.71	7.00E 14	1.71	7.04E 14	1.71	7.36E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.83E 13	1.71	3.40E 15	1.71	3.42E 15	1.71	3.57E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	7.98E 12	1.88	2.76E 14	1.88	2.78E 14	1.88	2.90E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.38E 04	1.09	1.17E 06	1.09	1.18E 06	1.09	1.23E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CORE STATION 0.2 FUEL ELEMENT TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 0.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.27E 02	1.14	8.70E 03	1.14	8.75E 03	1.14	9.17E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.45E 16	1.03	5.02E 17	1.03	5.05E 17	1.03	5.28E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.39E 16	1.03	2.21E 18	1.03	2.22E 18	1.03	2.32E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	2.00E 14	1.35	6.90E 15	1.35	6.94E 15	1.35	7.25E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.19E 07	1.14	8.44E 08	1.14	8.48E 08	1.14	8.89E 08	1.14

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.54E 02	1.14	3.77E 04	1.14	3.79E 04	1.14	3.97E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	7.84E 16	1.03	2.71E 18	1.03	2.73E 18	1.03	2.85E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.91E 17	1.03	1.35E 19	1.03	1.36E 19	1.03	1.42E 19	1.03
NEUTRONS/CM2, E LT 0.4EV	6.07E 13	1.35	2.10E 15	1.35	2.11E 15	1.35	2.20E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	9.46E 07	1.14	3.74E 09	1.14	3.76E 09	1.14	3.95E 09	1.14

TABLE 11-2 (Continued)

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 18.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.25E 02	1.14	3.65E 04	1.14	3.67E 04	1.14	3.85E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	7.67E 16	1.03	2.65E 18	1.03	2.67E 18	1.03	2.79E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.80E 17	1.03	1.32E 19	1.03	1.32E 19	1.03	1.38E 19	1.03
NEUTRONS/CM2, E LT 0.4EV	5.85E 13	1.35	2.02E 15	1.35	2.04E 15	1.35	2.13E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	9.16E 07	1.14	3.62E 09	1.14	3.64E 09	1.14	3.82E 09	1.14

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.94E 02	1.14	2.76E 04	1.14	2.77E 04	1.14	2.91E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	5.71E 16	1.03	1.98E 18	1.03	1.99E 18	1.03	2.08E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.37E 17	1.03	8.19E 18	1.03	8.24E 18	1.03	8.61E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	1.01E 14	1.35	3.51E 15	1.35	3.52E 15	1.35	3.68E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.88E 07	1.14	2.74E 09	1.14	2.75E 09	1.14	2.84E 09	1.14

TABLE 11-2 (Continued)

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 18.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.73E 02	1.14	2.67E 04	1.14	2.69E 04	1.14	2.82E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	5.59E 16	1.03	1.94E 18	1.03	1.95E 18	1.03	2.03E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.31E 17	1.03	7.98E 18	1.03	8.02E 18	1.03	8.38E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	9.76E 13	1.35	3.38E 15	1.35	3.39E 15	1.35	3.55E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.67E 07	1.14	2.65E 09	1.14	2.67E 09	1.14	2.80E 09	1.14

LOCATION - NSS-OUTLET-TO-TURBINE GAS TEMPERATURE
 RADIUS (IN) 7.30 STATION (IN) -35.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.67E-01	1.40	9.26E 00	1.40	9.31E 00	1.40	9.72E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.30E 13	3.10	4.51E 14	3.10	4.53E 14	3.10	4.74E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.12E 13	3.10	1.77E 15	3.10	1.78E 15	3.10	1.86E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.07E 13	3.10	3.72E 14	3.10	3.74E 14	3.10	3.90E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.34E 04	1.40	8.15E 05	1.40	8.19E 05	1.40	8.55E 05	1.40

TABLE 11-2 (Continued)

LOCATION - NSS INLET FROM TURBINE GAS TEMPERATURE
 RADIUS (IN) 3.00 STATION (IN) -32.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.11E-01	1.40	1.08E 01	1.40	1.09E 01	1.40	1.13E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.60E 13	3.10	5.52E 14	3.10	5.55E 14	3.10	5.80E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.90E 13	3.10	2.39E 15	3.10	2.40E 15	3.10	2.51E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.52E 13	3.10	5.26E 14	3.10	5.29E 14	3.10	5.53E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.74E 04	1.40	9.52E 05	1.40	9.57E 05	1.40	1.00E 06	1.40

LOCATION - FINAL SEAL EXIT GAS TEMPERATURE
 RADIUS (IN) 19.50 STATION (IN) 49.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.44E 01	1.14	3.68E 03	1.14	3.70E 03	1.14	3.88E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	7.45E 15	1.16	2.58E 17	1.16	2.59E 17	1.16	2.71E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.67E 16	1.16	1.62E 18	1.16	1.63E 18	1.16	1.70E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	1.54E 15	1.35	5.33E 16	1.35	5.36E 16	1.35	5.60E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	9.19E 06	1.14	3.59E 08	1.14	3.61E 08	1.14	3.79E 08	1.14

TABLE 11-2 (Continued)

LOCATION - CORE INLET PLENUM TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -3.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.33E 02	1.09	8.41E 03	1.09	8.45E 03	1.09	8.84E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.15E 16	1.71	3.99E 17	1.71	4.02E 17	1.71	4.20E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.43E 16	1.71	2.57E 18	1.71	2.58E 18	1.71	2.70E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	1.66E 15	1.88	5.75E 16	1.88	5.78E 16	1.88	6.05E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.15E 07	1.09	7.79E 08	1.09	7.83E 08	1.09	8.19E 08	1.09

LOCATION - CORE INLET PLENUM TEMPERATURE
 RADIUS (IN) 19.00 STATION (IN) -3.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.79E 01	1.09	3.53E 03	1.09	3.55E 03	1.09	3.71E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	4.94E 15	1.71	1.71E 17	1.71	1.72E 17	1.71	1.80E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.32E 16	1.71	1.15E 18	1.71	1.15E 18	1.71	1.21E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	1.76E 15	1.88	6.07E 16	1.88	6.10E 16	1.88	6.38E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	8.90E 06	1.09	3.23E 08	1.09	3.25E 08	1.09	3.40E 08	1.09

TABLE 11-2 (Continued)

LOCATION - REFLECTOR INLET PLINUM TEMPERATURE
 RADIUS (IN) 24.50 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.00E 01	1.40	7.42E 02	1.40	7.45E 02	1.40	7.80E 02	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.48E 14	3.10	2.93E 16	3.10	2.95E 16	3.10	3.08E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.88E 15	3.10	2.04E 17	3.10	2.05E 17	3.10	2.14E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.14E 15	3.10	3.95E 16	3.10	3.97E 16	3.10	4.15E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.85E 06	1.40	6.90E 07	1.40	6.94E 07	1.40	7.27E 07	1.40

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)
 NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.20E 01	1.14	1.55E 03	1.14	1.56E 03	1.14	1.63E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	9.96E 14	1.16	3.45E 16	1.16	3.46E 16	1.16	3.62E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.18E 16	1.16	4.09E 17	1.16	4.11E 17	1.16	4.30E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.24E 15	1.35	7.74E 16	1.35	7.78E 16	1.35	8.13E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	3.83E 06	1.14	1.43E 08	1.14	1.43E 08	1.14	1.50E 08	1.14

TABLE 11-2 (Continued)

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	7.57E 01	1.14	2.83E 03	1.14	2.84E 03	1.14	2.97E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.95E 15	1.16	6.73E 16	1.16	6.77E 16	1.16	7.07E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.42E 16	1.16	8.37E 17	1.16	8.41E 17	1.16	8.79E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	4.62E 15	1.35	1.60E 17	1.35	1.61E 17	1.35	1.68E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.96E 06	1.14	2.62E 08	1.14	2.63E 08	1.14	2.76E 08	1.14

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 48.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.81E 01	1.14	1.05E 03	1.14	1.05E 03	1.14	1.10E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	6.77E 14	1.16	2.34E 16	1.16	2.35E 16	1.16	2.46E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.96E 15	1.16	2.76E 17	1.16	2.77E 17	1.16	2.89E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	1.50E 15	1.35	5.18E 16	1.35	5.20E 16	1.35	5.44E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.58E 06	1.14	9.69E 07	1.14	9.74E 07	1.14	1.02E 08	1.14

TABLE 11-2 (Continued)

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 22.20 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.74E 02	1.14	6.63E 03	1.14	6.66E 03	1.14	6.98E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.18E 16	1.16	4.07E 17	1.16	4.09E 17	1.16	4.28E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.28E 17	1.16	4.44E 18	1.16	4.46E 18	1.16	4.67E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	8.41E 15	1.35	2.91E 17	1.35	2.93E 17	1.35	3.06E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.65E 07	1.14	6.33E 08	1.14	6.36E 08	1.14	6.67E 08	1.14

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 22.20 STATION (IN) 4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.29E 01	1.14	3.50E 03	1.14	3.52E 03	1.14	3.69E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	6.00E 15	1.16	2.08E 17	1.16	2.09E 17	1.16	2.18E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.26E 16	1.16	2.17E 18	1.16	2.18E 18	1.16	2.28E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	4.19E 15	1.35	1.45E 17	1.35	1.46E 17	1.35	1.52E 17	1.35
GAMMA RAY DOSE, RADS (CARBON)	8.75E 06	1.14	3.32E 08	1.14	3.34E 08	1.14	3.50E 08	1.14

11-41

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM GAS TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -5.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.55E 02	1.09	5.54E 03	1.09	5.57E 03	1.09	5.83E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	7.45E 15	1.71	2.58E 17	1.71	2.59E 17	1.71	2.71E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.78E 16	1.71	2.00E 18	1.71	2.01E 18	1.71	2.10E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	3.90E 15	1.88	1.35E 17	1.88	1.36E 17	1.88	1.42E 17	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.41E 07	1.09	5.05E 08	1.09	5.08E 08	1.09	5.31E 08	1.09

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 22.20 STATION (IN) 48.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.11E 01	1.14	2.32E 03	1.14	2.33E 03	1.14	2.44E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	3.99E 15	1.16	1.38E 17	1.16	1.39E 17	1.16	1.45E 17	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.15E 16	1.16	1.43E 18	1.16	1.44E 18	1.16	1.51E 18	1.16
NEUTRONS/CM2, E LT 0.4EV	2.72E 15	1.35	9.41E 16	1.35	9.46E 16	1.35	9.89E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	5.79E 06	1.14	2.21E 08	1.14	2.22E 08	1.14	2.33E 08	1.14

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM GAS TEMPERATURE
 RADIUS (IN) 17.00 STATION (IN) -5.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.59E 01	1.09	3.06E 03	1.09	3.08E 03	1.09	3.22E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	4.02E 15	1.71	1.39E 17	1.71	1.40E 17	1.71	1.46E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.06E 16	1.71	1.06E 18	1.71	1.06E 18	1.71	1.11E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	2.44E 15	1.88	8.43E 16	1.88	8.44E 16	1.88	8.86E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	7.73E 06	1.09	2.77E 08	1.09	2.78E 08	1.09	2.91E 08	1.09

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.53E 02	1.14	2.57E 04	1.14	2.59E 04	1.14	2.71E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	5.53E 16	1.03	1.91E 18	1.03	1.92E 18	1.03	2.01E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.67E 17	1.03	9.23E 18	1.03	9.28E 18	1.03	9.70E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	4.46E 13	1.35	1.54E 15	1.35	1.55E 15	1.35	1.62E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.46E 07	1.14	2.55E 09	1.14	2.56E 09	1.14	2.69E 09	1.14

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.54E 02	1.14	3.77E 04	1.14	3.79E 04	1.14	3.97E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	7.84E 16	1.03	2.71E 18	1.03	2.73E 18	1.03	2.85E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.91E 17	1.03	1.35E 19	1.03	1.36E 19	1.03	1.42E 19	1.03
NEUTRONS/CM2, E LT 0.4EV	6.07E 13	1.35	2.10E 15	1.35	2.11E 15	1.35	2.20E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	9.46E 07	1.14	3.74E 09	1.14	3.76E 09	1.14	3.95E 09	1.14

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 50.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.27E 02	1.14	8.91E 03	1.14	8.96E 03	1.14	9.40E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.86E 16	1.03	6.43E 17	1.03	6.46E 17	1.03	6.75E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.73E 16	1.03	3.02E 18	1.03	3.03E 18	1.03	3.17E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	1.56E 13	1.35	5.41E 14	1.35	5.44E 14	1.35	5.68E 14	1.35
GAMMA RAY DOSE, RADS (CARBON)	2.24E 07	1.14	8.82E 08	1.14	8.86E 08	1.14	9.30E 08	1.14

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.85E 02	1.14	2.72E 04	1.14	2.74E 04	1.14	2.87E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	5.60E 16	1.03	1.94E 18	1.03	1.95E 18	1.03	2.03E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.32E 17	1.03	8.04E 18	1.03	8.08E 18	1.03	8.44E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	1.24E 14	1.35	4.29E 15	1.35	4.31E 15	1.35	4.51E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	6.79E 07	1.14	2.70E 09	1.14	2.72E 09	1.14	2.85E 09	1.14

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.68E 02	1.14	1.85E 04	1.14	1.86E 04	1.14	1.96E 04	1.14
NEUTRONS/CM2, E GT 1.0MEV	3.97E 16	1.03	1.37E 18	1.03	1.38E 18	1.03	1.44E 18	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.59E 17	1.03	5.52E 18	1.03	5.55E 18	1.03	5.80E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	8.37E 13	1.35	2.90E 15	1.35	2.91E 15	1.35	3.04E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	4.63E 07	1.14	1.84E 09	1.14	1.85E 09	1.14	1.94E 09	1.14

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT EXHAUST PLENUM TEMPERATURE
 RADIUS (IN) 21.50 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.51E 01	1.09	1.59E 03	1.09	1.60E 03	1.09	1.67E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.57E 15	1.71	5.43E 16	1.71	5.46E 16	1.71	5.70E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.33E 16	1.71	4.59E 17	1.71	4.62E 17	1.71	4.83E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	1.93E 15	1.88	6.67E 16	1.88	6.70E 16	1.88	7.01E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.97E 06	1.09	1.41E 08	1.09	1.42E 08	1.09	1.48E 08	1.09

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 50.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.61E 02	1.14	6.37E 03	1.14	6.41E 03	1.14	6.72E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.32E 16	1.03	4.55E 17	1.03	4.53E 17	1.03	4.78E 17	1.03
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.21E 16	1.03	1.80E 18	1.03	1.81E 18	1.03	1.89E 18	1.03
NEUTRONS/CM2, E LT 0.4EV	3.07E 13	1.35	1.06E 15	1.35	1.07E 15	1.35	1.12E 15	1.35
GAMMA RAY DOSE, RADS (CARBON)	1.59E 07	1.14	6.31E 08	1.14	6.34E 08	1.14	6.65E 08	1.14

TABLE 11-2 (Continued)

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 8.00 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.14E 02	1.09	4.05E 03	1.09	4.07E 03	1.09	4.25E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.20E 15	1.71	1.80E 17	1.71	1.81E 17	1.71	1.89E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.22E 16	1.71	1.46E 18	1.71	1.47E 18	1.71	1.53E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	3.71E 15	1.88	1.28E 17	1.88	1.29E 17	1.88	1.35E 17	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.02E 07	1.09	3.65E 08	1.09	3.67E 08	1.09	3.64E 08	1.09

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.28E 02	1.09	4.54E 03	1.09	4.57E 03	1.09	4.77E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	5.74E 15	1.71	1.99E 17	1.71	2.00E 17	1.71	2.09E 17	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.74E 16	1.71	1.64E 18	1.71	1.65E 18	1.71	1.72E 18	1.71
NEUTRONS/CM2, E LT 0.4EV	4.19E 15	1.88	1.45E 17	1.88	1.46E 17	1.88	1.52E 17	1.88
GAMMA RAY DOSE, RADS (CARBON)	1.15E 07	1.09	4.10E 08	1.09	4.12E 08	1.09	4.31E 08	1.09

TABLE 11-2 (Continued)

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 28.08 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.27E 01	1.40	1.60E 03	1.40	1.61E 03	1.40	1.68E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.94E 14	3.10	3.09E 16	3.10	3.11E 16	3.10	3.25E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.40E 15	3.10	2.56E 17	3.10	2.57E 17	3.10	2.69E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.94E 16	3.10	4.97E 16	3.10	5.19E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.93E 06	1.40	1.48E 08	1.40	1.49E 08	1.40	1.56E 08	1.40

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 19.00 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.00E 01	1.09	2.12E 03	1.09	2.13E 03	1.09	2.23E 03	1.09
NEUTRONS/CM2, E GT 1.0MEV	2.43E 15	1.71	8.42E 16	1.71	8.46E 16	1.71	8.84E 16	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.01E 16	1.71	6.94E 17	1.71	6.98E 17	1.71	7.29E 17	1.71
NEUTRONS/CM2, E LT 0.4EV	2.27E 15	1.88	7.85E 16	1.88	7.89E 16	1.88	8.25E 16	1.88
GAMMA RAY DOSE, RADS (CARBON)	5.32E 06	1.09	1.89E 08	1.09	1.90E 08	1.09	1.99E 08	1.09

TABLE 11-2 (Continued)

LOCATION - REFLECTOR INLET/TURBINE INLET AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	2.42E-01	1.09	8.35E 00	1.09	8.40E 00	1.09	8.77E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.34E 13	1.71	4.65E 14	1.71	4.67E 14	1.71	4.88E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.43E 13	1.71	2.23E 15	1.71	2.24E 15	1.71	2.34E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.09E 13	1.88	3.77E 14	1.88	3.79E 14	1.88	3.97E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.11E 04	1.09	7.31E 05	1.09	7.35E 05	1.09	7.68E 05	1.09

LOCATION - REFLECTOR INLET/TURBINE INLET AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.90

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	3.24E-01	1.40	1.12E 01	1.40	1.13E 01	1.40	1.18E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.84E 13	3.10	6.37E 14	3.10	6.40E 14	3.10	6.69E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.99E 13	3.10	2.77E 15	3.10	2.78E 15	3.10	2.91E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.48E 13	3.10	5.11E 14	3.10	5.14E 14	3.10	5.37E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.85E 04	1.40	9.91E 05	1.40	9.95E 05	1.40	1.04E 06	1.40

TABLE 11-2 (Continued)

LOCATION - SEAL CHAMBER TO CORE EXIT DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTR NS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTR NS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTR NS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - SEAL CHAMBER TO CORE EXIT DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.92E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT SYSTEM DIFFERENTIAL PRESSURE (SSCA INLET TO REFL OUT)
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - STRUCTURAL SUPPORT SYSTEM DIFFERENTIAL PRESSURE (SSCA INLET TO REFL OUT)
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2(Continued)

LOCATION - REFLECTOR/INLET CORE OUTLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - REFLECTOR/INLET CORE OUTLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - REFLECTOR INLET PLENUM PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.42E-01	1.09	8.35E 00	1.09	8.40E 00	1.09	8.77E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.34E 13	1.71	4.65E 14	1.71	4.67E 14	1.71	4.88E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.43E 13	1.71	2.23E 15	1.71	2.24E 15	1.71	2.34E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.09E 13	1.88	3.77E 14	1.88	3.79E 14	1.88	3.97E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.11E 04	1.09	7.31E 05	1.09	7.35E 05	1.09	7.68E 05	1.09

LOCATION - REFLECTOR INLET PLENUM PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.24E-01	1.40	1.12E 01	1.40	1.13E 01	1.40	1.18E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.44E 13	3.10	6.37E 14	3.10	6.40E 14	3.10	6.69E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.99E 13	3.10	2.77E 15	3.10	2.79E 15	3.10	2.91E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.48E 13	3.10	5.11E 14	3.10	5.14E 14	3.10	5.37E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.85E 04	1.40	9.91E 05	1.40	9.95E 05	1.40	1.04E 06	1.40

TABLE 11-2 (Continued)

LOCATION - NSS INLET FROM TURBINE PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.24E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - NSS INLET FROM TURBINE PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.42E-01	1.09	8.35E 00	1.09	8.40E 00	1.09	8.77E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.34E 13	1.71	4.65E 14	1.71	4.67E 14	1.71	4.88E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.43E 13	1.71	2.23E 15	1.71	2.24E 15	1.71	2.34E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.09E 13	1.88	3.77E 14	1.88	3.79E 14	1.88	3.97E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.11E 04	1.09	7.31E 05	1.09	7.35E 05	1.09	7.68E 05	1.09

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.24E-01	1.40	1.12E 01	1.40	1.13E 01	1.40	1.18E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.84E 13	3.10	6.37E 14	3.10	6.40E 14	3.10	6.69E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.99E 13	3.10	2.77E 15	3.10	2.78E 15	3.10	2.91E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.48E 13	3.10	5.11E 14	3.10	5.14E 14	3.10	5.37E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.85E 04	1.40	9.91E 05	1.40	9.95E 05	1.40	1.04E 06	1.40

TABLE 11-2 (Continued)

LOCATION - CORE AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - CORE AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - NSS OUTLET TO TURBINE PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.24E-01	1.40	1.12E 01	1.40	1.13E 01	1.40	1.18E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.84E 13	3.10	6.37E 14	3.10	6.40E 14	3.10	6.69E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.99E 13	3.10	2.77E 15	3.10	2.78E 15	3.10	2.91E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.48E 13	3.10	5.11E 14	3.10	5.14E 14	3.10	5.37E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.85E 04	1.40	9.91E 05	1.40	9.95E 05	1.40	1.04E 06	1.40

LOCATION - NSS OUTLET TO TURBINE PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.42E-01	1.09	8.35E 00	1.09	8.40E 00	1.09	8.77E 00	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.34E 13	1.71	4.65E 14	1.71	4.67E 14	1.71	4.88E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.43E 13	1.71	2.23E 15	1.71	2.24E 15	1.71	2.34E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.09E 13	1.88	3.77E 14	1.88	3.79E 14	1.88	3.97E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	2.11E 04	1.09	7.31E 05	1.09	7.35E 05	1.09	7.68E 05	1.09

TABLE 11-2 (Continued)

LOCATION - NSS INLET FROM TURBINE TO CORE INLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - NSS INLET FROM TURBINE TO CORE INLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.63E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - FAST NEUTRON FLUX
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.75E-01	1.09	1.30E 01	1.09	1.30E 01	1.09	1.36E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	1.95E 13	1.71	6.76E 14	1.71	6.80E 14	1.71	7.11E 14	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.52E 13	1.71	3.29E 15	1.71	3.31E 15	1.71	3.46E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	8.34E 12	1.88	2.88E 14	1.88	2.90E 14	1.88	3.03E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.26E 04	1.09	1.13E 06	1.09	1.14E 06	1.09	1.19E 06	1.09

LOCATION - FAST NEUTRON FLUX
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.52E-01	1.09	1.57E 01	1.09	1.58E 01	1.09	1.65E 01	1.09
NEUTRONS/CM2, E GT 1.0MEV	3.52E 13	1.71	1.22E 15	1.71	1.23E 15	1.71	1.28E 15	1.71
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.41E 14	1.71	4.89E 15	1.71	4.91E 15	1.71	5.13E 15	1.71
NEUTRONS/CM2, E LT 0.4EV	1.27E 13	1.88	4.39E 14	1.88	4.41E 14	1.88	4.61E 14	1.88
GAMMA RAY DOSE, RADS (CARBON)	3.94E 04	1.09	1.37E 06	1.09	1.38E 06	1.09	1.44E 06	1.09

TABLE 11-2 (Continued)

LOCATION - CORE NUCLEAR POWER (INTERMEDIATE RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	4.87E 01	1.14	1.74E 03	1.14	1.75E 03	1.14	1.83E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.75E 15	1.16	6.05E 16	1.16	6.08E 16	1.16	6.36E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.48E 16	1.16	5.13E 17	1.16	5.15E 17	1.16	5.39E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.20E 15	1.35	7.60E 16	1.35	7.63E 16	1.35	7.98E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	4.33E 06	1.14	1.56E 08	1.14	1.57E 08	1.14	1.64E 08	1.14

LOCATION - CORE NUCLEAR POWER (INTERMEDIATE RANGE)
 RADIUS (IN) 28.00 STATION (IN) 26.00

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	4.27E 01	1.40	1.60E 03	1.40	1.61E 03	1.40	1.68E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.94E 14	3.10	3.09E 16	3.10	3.11E 16	3.10	3.25E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.40E 15	3.10	2.56E 17	3.10	2.57E 17	3.10	2.69E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.94E 16	3.10	4.97E 16	3.10	5.19E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.93E 06	1.40	1.48E 08	1.40	1.49E 08	1.40	1.56E 08	1.40

TABLE 11-2 (Continued)

LOCATION - CORE NUCLEAR POWER(HIGH RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (F)

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WATT-SEC/GM-304 STAINLESS STEEL	4.87E 01	1.14	1.74E 03	1.14	1.75E 03	1.14	1.83E 03	1.14
NEUTRONS/CM2, E GT 1.0MEV	1.75E 15	1.16	6.05E 16	1.16	6.08E 16	1.16	6.36E 16	1.16
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.48E 16	1.16	5.13E 17	1.16	5.15E 17	1.16	5.39E 17	1.16
NEUTRONS/CM2, E LT 0.4EV	2.20E 15	1.35	7.60E 16	1.35	7.63E 16	1.35	7.98E 16	1.35
GAMMA RAY DOSE, RADS (CARBON)	4.33E 06	1.14	1.56E 08	1.14	1.57E 08	1.14	1.64E 08	1.14

LOCATION - CORE NUCLEAR POWER(HIGH RANGE)
 RADIUS (IN) 28.00 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.27E 01	1.40	1.60E 03	1.40	1.61E 03	1.40	1.68E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.94E 14	3.10	3.09E 16	3.10	3.11E 16	3.10	3.25E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.40E 15	3.10	2.56E 17	3.10	2.57E 17	3.10	2.69E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.94E 16	3.10	4.97E 16	3.10	5.19E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	3.93E 06	1.40	1.48E 08	1.40	1.49E 08	1.40	1.56E 08	1.40

TABLE 11-2 (Continued)

LOCATION - SSCV POSITION
 RADIUS (IN) 42.50 STATION (IN) -111.05

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.85E-02	1.40	2.03E 00	1.40	2.05E 00	1.40	2.14E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.60E 11	3.10	1.59E 13	3.10	1.60E 13	3.10	1.67E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.10E 12	3.10	1.07E 14	3.10	1.08E 14	3.10	1.13E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	5.60E 11	3.10	1.94E 13	3.10	1.95E 13	3.10	2.04E 13	3.10
GAMMA RAY DOSE, RAD5 (CARBON)	4.46E 03	1.40	1.55E 05	1.40	1.56E 05	1.40	1.63E 05	1.40

LOCATION - SSCV POSITION
 RADIUS (IN) 49.00 STATION (IN) -103.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.03E-01	1.40	3.60E 00	1.40	3.62E 00	1.40	3.78E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.47E 11	3.10	1.55E 13	3.10	1.55E 13	3.10	1.62E 13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.91E 12	3.10	1.35E 14	3.10	1.36E 14	3.10	1.42E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	7.12E 11	3.10	2.46E 13	3.10	2.48E 13	3.10	2.59E 13	3.10
GAMMA RAY DOSE, RAD5 (CARBON)	7.89E 03	1.40	2.75E 05	1.40	2.76E 05	1.40	2.89E 05	1.40

TABLE 11-3

MAXIMUM NOMINAL RADIATION ENVIRONMENT AT THE INSTRUMENTATION LOCATIONS
FOR THE COMPOSITE AND GRAPHITE CORES (DWG. 939J723)

	Neutron Environment			Photon Environment	
	$E < 0.4 \text{ eV}$ $n/\text{cm}^2\text{-sec}$	$0.4 \text{ eV} < E < 1.0 \text{ MeV}$ $n/\text{cm}^2\text{-sec}$	$E > 1.0 \text{ MeV}$ $n/\text{cm}^2\text{-sec}$	$\frac{\text{Rads-C}}{\text{HR}}$	$\frac{\text{Watts}}{\text{Gm(SS)}}$
Composite Core	5.11(14)	2.37(16)	4.76(15)	2.46(10)	6.88(1)
Graphite Core	5.45(14)	2.29(16)	4.51(15)	3.39(10)	9.45(1)

NOTE: Numbers in parentheses () refer to powers of ten, i. e., $5.11(14) \equiv 5.11 \times 10^{14}$

TABLE 11-4

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE GRAPHITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E<0.4 eV			0.4 eV<E<1.0 MeV			E>1.0 MeV		
		Radius (In)	Station (In)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
1	Core Station 0.2 Fuel Element Temp.	0.0	0.2	2.94 (1)	1.03 (10)	1.03 (10)	1.03 (11)	2.39 (13)	8.61 (16)	8.61 (17)	6.04 (15)	2.18 (19)	2.18 (20)	1.15 (15)	4.16 (18)	4.16 (19)
		15.75	0.2	1.90 (1)	6.66 (9)	6.66 (9)	6.66 (10)	1.50 (13)	5.40 (16)	5.40 (17)	3.67 (15)	1.32 (19)	1.32 (20)	8.40 (14)	3.03 (18)	3.03 (19)
2	Core Unfueled Element Temp.	0.0	18.0	9.16 (1)	3.28 (10)	3.28 (10)	3.28 (11)	1.76 (13)	6.33 (16)	6.33 (17)	2.24 (16)	8.05 (19)	8.05 (20)	4.41 (15)	1.59 (19)	1.59 (20)
		0.0	26.0	9.45 (1)	3.39 (10)	3.39 (10)	3.39 (11)	1.82 (13)	6.54 (16)	6.54 (17)	2.29 (16)	8.26 (19)	8.26 (20)	4.51 (15)	1.62 (19)	1.62 (20)
		15.75	18.0	6.18 (1)	2.21 (10)	2.21 (10)	2.21 (11)	1.22 (13)	4.41 (16)	4.41 (17)	1.36 (16)	4.91 (19)	4.91 (20)	3.28 (15)	1.18 (19)	1.18 (20)
		15.75	26.0	6.37 (1)	2.28 (10)	2.28 (10)	2.28 (11)	1.27 (13)	4.57 (16)	4.57 (17)	1.40 (16)	5.03 (19)	5.03 (20)	3.35 (15)	1.21 (19)	1.21 (20)
3	Cluster Exit Gas Temperature	0.0	55.0	5.37 (0)	1.86 (9)	1.86 (9)	1.86 (10)	4.92 (12)	1.77 (16)	1.77 (17)	1.02 (15)	3.68 (18)	3.68 (19)	3.08 (14)	1.11 (18)	1.11 (19)
		15.75	55.0	3.34 (0)	1.15 (9)	1.15 (9)	1.15 (10)	1.13 (13)	4.23 (16)	4.23 (17)	7.06 (14)	2.54 (18)	2.54 (19)	2.03 (14)	7.29 (17)	7.29 (18)
4	Structural Support Supply Plenum Gas Temperature	0.0	-5.2	1.10 (1)	3.66 (9)	3.66 (9)	3.66 (10)	2.34 (14)	8.43 (17)	8.43 (18)	3.13 (15)	1.13 (19)	1.13 (20)	4.24 (14)	1.53 (18)	1.53 (19)
		17.0	-5.2	5.92 (0)	1.95 (9)	1.95 (9)	1.95 (10)	1.46 (14)	5.24 (17)	5.24 (18)	1.68 (15)	6.05 (18)	6.05 (19)	2.31 (14)	8.33 (17)	8.33 (18)
5	Structural Support Liner Temperature	0.0	8.0	6.43 (1)	2.30 (10)	2.30 (10)	2.30 (11)	1.25 (13)	4.51 (16)	4.51 (17)	1.57 (16)	5.66 (19)	5.66 (20)	3.18 (15)	1.15 (19)	1.15 (20)
		0.0	26.0	9.45 (1)	3.39 (10)	3.39 (10)	3.39 (11)	1.82 (13)	6.54 (16)	6.54 (17)	2.29 (16)	8.26 (19)	8.26 (20)	4.51 (15)	1.62 (19)	1.62 (20)
		0.0	50.0	2.27 (1)	8.11 (9)	8.11 (9)	8.11 (10)	4.07 (12)	1.47 (16)	1.47 (17)	5.05 (15)	1.82 (19)	1.82 (20)	1.05 (15)	3.80 (18)	3.80 (19)
		16.0	8.0	4.26 (1)	1.53 (10)	1.53 (10)	1.53 (11)	9.89 (12)	3.56 (16)	3.56 (17)	9.44 (15)	3.40 (19)	3.40 (20)	2.33 (15)	8.38 (18)	8.38 (19)
		16.0	26.0	6.26 (1)	2.25 (10)	2.25 (10)	2.25 (11)	1.47 (13)	5.29 (16)	5.29 (17)	1.37 (16)	4.93 (19)	4.93 (20)	3.28 (15)	1.18 (19)	1.18 (20)
		16.0	50.0	1.48 (1)	5.27 (9)	5.27 (9)	5.27 (10)	3.46 (12)	1.25 (16)	1.25 (17)	3.04 (15)	1.09 (19)	1.09 (20)	7.65 (14)	2.76 (18)	2.76 (19)
6	Deleted**															
7	Deleted**															
8	Structural Support Exhaust Plenum Temp.	21.5	-6.5	3.01 (0)	9.73 (8)	9.73 (8)	9.73 (9)	1.14 (14)	4.11 (17)	4.11 (18)	7.32 (14)	2.63 (18)	2.63 (19)	9.06 (14)	3.26 (17)	3.26 (18)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference 3

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91 (-3) $\equiv$ 8.91 $\times$ 10⁻³
2) Data presented for 1511 MW thermal power.

579	10/8/70	471	10/12/70
580	10/8/70	450	10/12/70
575	10/8/70	429	10/12/70

TABLE 11-4
(Continued)RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR THE GRAPHITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E<0.4 eV			0.4 eV<E<1.0 MeV			E>1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-G/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
9	NSS-Inlet-From-Turbine Gas Temp.	3.0	-32.5	1.95 (-2)	6.23 (6)	6.23 (6)	6.23 (7)	8.64 (11)	3.11 (15)	3.11 (16)	3.94 (12)	1.42 (16)	1.42 (17)	9.40 (11)	3.38 (15)	3.38 (16)
10	Reflector Inlet Plenum Temperature	24.5	55.0	1.61 (0)	5.46 (8)	5.46 (8)	5.46 (9)	6.87 (13)	2.47 (17)	2.47 (18)	3.27 (14)	1.18 (18)	1.18 (19)	4.87 (13)	1.75 (17)	1.75 (18)
11	NSS-Outlet-To-Turbine Gas Temp.	7.3	-35.0	1.67 (-2)	5.31 (6)	5.31 (6)	5.31 (7)	6.10 (11)	2.20 (15)	2.20 (16)	2.93 (12)	1.05 (16)	1.05 (17)	7.68 (11)	2.76 (15)	2.76 (16)
12	Reflector Material Temperature	26.5	4.0	3.24 (0)	1.09 (9)	1.09 (9)	1.09 (10)	1.35 (14)	4.88 (17)	4.88 (18)	6.66 (14)	2.40 (19)	2.40 (20)	5.89 (13)	2.12 (17)	2.12 (18)
		26.5	26.0	5.97 (0)	2.02 (9)	2.02 (9)	2.02 (10)	2.81 (14)	1.01 (18)	1.01 (19)	1.36 (15)	4.90 (18)	4.90 (19)	1.15 (14)	4.14 (17)	4.14 (18)
		26.5	48.0	2.24 (0)	7.55 (8)	7.55 (8)	7.55 (9)	9.07 (13)	3.27 (17)	3.27 (18)	4.47 (14)	1.61 (18)	1.61 (19)	3.99 (13)	1.44 (17)	1.44 (18)
		22.2	4.0	7.61 (0)	2.62 (9)	2.62 (9)	2.62 (10)	2.68 (14)	9.64 (17)	9.64 (18)	3.58 (15)	1.29 (19)	1.29 (20)	3.51 (14)	1.26 (18)	1.26 (19)
		22.2	26.0	1.45 (1)	5.04 (9)	5.04 (9)	5.04 (10)	5.45 (14)	1.96 (18)	1.96 (19)	7.34 (15)	2.64 (19)	2.64 (20)	6.88 (14)	2.48 (18)	2.48 (19)
		22.2	48.0	5.15 (0)	1.78 (9)	1.78 (9)	1.78 (10)	1.75 (14)	6.29 (17)	6.29 (18)	2.36 (15)	8.51 (18)	8.51 (19)	2.32 (14)	8.36 (17)	8.36 (18)
13	Final Seal Exit Gas Temperature	19.5	49.0	8.50 (0)	3.00 (9)	3.00 (9)	3.00 (10)	1.03 (14)	3.71 (17)	3.71 (18)	2.68 (15)	9.64 (18)	9.64 (19)	4.31 (14)	1.55 (18)	1.55 (19)
14	Central Shield Material Temp.	0.0	-11.83	3.65 (0)	1.21 (9)	1.21 (9)	1.21 (10)	3.77 (13)	1.36 (17)	1.36 (18)	8.92 (14)	3.21 (18)	3.21 (19)	1.13 (14)	4.08 (17)	4.08 (18)
		0.0	-17.5	7.56 (-1)	2.52 (8)	2.52 (8)	2.52 (9)	1.15 (11)	4.15 (14)	4.15 (15)	7.73 (13)	2.78 (17)	2.78 (18)	1.70 (13)	6.12 (16)	6.12 (17)
		0.0	-22.93	1.51 (-1)	4.89 (7)	4.89 (7)	4.89 (8)	1.83 (11)	6.58 (14)	6.58 (15)	1.21 (13)	4.36 (16)	4.36 (17)	3.22 (12)	1.16 (16)	1.16 (17)
		0.0	-24.23	2.88 (-2)	9.15 (6)	9.15 (6)	9.15 (7)	6.88 (11)	2.48 (15)	2.48 (16)	8.18 (12)	2.94 (16)	2.94 (17)	2.11 (12)	7.60 (15)	7.60 (16)
		10.0	-11.83	3.03 (0)	1.01 (9)	1.01 (9)	1.01 (10)	3.12 (13)	1.12 (17)	1.12 (18)	7.38 (14)	2.66 (18)	2.66 (19)	9.49 (13)	3.42 (17)	3.42 (18)
		10.0	-17.5	6.32 (-1)	2.10 (8)	2.10 (8)	2.10 (9)	9.57 (10)	3.44 (14)	3.44 (15)	6.43 (13)	2.32 (17)	2.32 (18)	1.42 (13)	5.13 (16)	5.13 (17)
		10.0	-22.93	1.27 (-1)	4.12 (7)	4.12 (7)	4.12 (8)	1.67 (11)	6.00 (14)	6.00 (15)	1.03 (13)	3.69 (16)	3.69 (17)	2.79 (12)	1.01 (16)	1.01 (17)
		10.0	-24.23	2.48 (-2)	7.86 (6)	7.86 (6)	7.86 (7)	6.30 (11)	2.27 (15)	2.27 (16)	7.09 (12)	2.55 (16)	2.55 (17)	1.85 (12)	6.64 (15)	6.64 (16)
		19.4	-11.83	1.59 (0)	5.23 (8)	5.23 (8)	5.23 (9)	1.95 (13)	7.02 (16)	7.02 (17)	3.73 (14)	1.34 (18)	1.34 (19)	4.98 (13)	1.79 (17)	1.79 (18)
		19.4	-17.5	3.50 (-1)	1.16 (8)	1.16 (8)	1.16 (9)	1.34 (11)	4.82 (14)	4.82 (15)	4.33 (13)	1.56 (17)	1.56 (18)	8.39 (12)	3.02 (16)	3.02 (17)
		19.4	-22.93	7.70 (-2)	2.47 (7)	2.47 (7)	2.47 (8)	1.80 (11)	6.46 (14)	6.46 (15)	7.74 (12)	2.79 (16)	2.79 (17)	1.73 (12)	6.22 (15)	6.22 (16)
		19.4	-24.23	2.42 (-2)	7.66 (6)	7.66 (6)	7.66 (7)	4.53 (11)	1.63 (15)	1.63 (16)	5.61 (12)	2.02 (16)	2.02 (17)	1.19 (12)	4.29 (15)	4.29 (16)

*Consistent with designation in Table 1 of EC-677565.

*Per Reference 3.

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91 (-3) = 8.91×10^{-3}
2) Data presented for 1511 MW thermal power.

TABLE 11-4
(Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR GRAPHITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E<0.4 eV			0.4 eV<E<1.0 MeV			E>1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
15	Core Inlet Plenum Temperature	0.0 19.0	- 3.0 - 3.0	1.71 (1) 7.07 (0)	5.79 (9) 2.36 (9)	5.79 (9) 2.36 (9)	5.79(10) 2.36 (10)	1.02 (14) 1.07 (14)	3.68 (17) 3.84 (17)	3.68 (18) 3.84 (18)	4.06 (15) 1.84 (15)	1.46 (19) 6.62 (18)	1.46 (20) 6.62 (19)	6.54 (14) 2.83 (14)	2.36 (18) 1.02 (18)	2.36 (19) 1.02 (19)
16	Support Plate Material Temp.	0.0 8.0 19.0	- 6.5 - 6.5 - 6.5	8.79 (0) 7.81 (0) 4.02 (0)	2.90 (9) 2.57 (9) 1.31 (9)	2.90 (9) 2.57 (9) 1.31 (9)	2.90 (10) 2.57 (10) 1.31 (10)	2.49 (14) 2.20 (14) 1.35 (14)	8.97 (17) 7.94 (17) 4.85 (17)	8.97 (18) 7.94 (18) 4.85 (18)	2.56 (15) 2.29 (15) 1.10 (15)	9.23 (18) 8.25 (18) 3.97 (18)	9.23 (19) 8.25 (19) 3.97 (19)	3.28 (14) 2.98 (14) 1.40 (14)	1.18 (18) 1.07 (18) 5.04 (17)	1.18 (19) 1.07 (19) 5.04 (18)
17	Nuclear Subsystem Vibration	0.0 22.0	- 9.5 - 7.0	5.23 (0) 2.60 (0)	1.70 (9) 8.36 (8)	1.70 (9) 8.36 (8)	1.70 (10) 8.36 (9)	1.54 (14) 1.04 (14)	5.54 (17) 3.73 (17)	5.54 (18) 3.73 (18)	1.40 (15) 6.05 (14)	5.05 (18) 2.18 (18)	5.05 (19) 2.18 (19)	1.69 (14) 7.23 (13)	6.08 (17) 2.62 (17)	6.08 (17) 2.62 (18)
18	Deleted**															
19	Deleted**															
20	Structural Support System Load	0.0 15.75	- 8.0 - 8.0	6.89 (0) 4.18 (0)	2.25 (9) 1.36 (9)	2.25 (9) 1.36 (9)	2.25 (10) 1.36 (10)	2.22 (14) 1.40 (14)	8.01 (17) 5.04 (17)	8.01 (18) 5.04 (18)	1.98 (15) 1.17 (15)	7.13 (18) 4.23 (18)	7.13 (19) 4.23 (19)	2.41 (14) 1.54 (14)	8.67 (17) 5.21 (17)	8.67 (18) 5.21 (18)
21	Reflector Inlet Plenum Pressure	7.0 19.5	-30.8 -28.8	2.03 (-2) 1.49 (-2)	6.48 (6) 4.73 (6)	6.48 (6) 4.73 (6)	6.48 (7) 4.73 (7)	8.39 (11) 6.19 (11)	3.02 (15) 2.23 (15)	3.02 (16) 2.23 (16)	4.56 (12) 3.68 (12)	1.64 (16) 1.32 (16)	1.64 (17) 1.32 (17)	1.08 (12) 7.91 (11)	3.90 (15) 2.85 (15)	3.90 (16) 2.85 (16)
22	Deleted**															
23	Deleted**															
24	NSS Outlet To Turbine Pressure	7.0 19.5	-30.8 -28.8	2.03 (-2) 1.49 (-2)	6.48 (6) 4.73 (6)	6.48 (6) 4.73 (6)	6.48 (7) 4.73 (7)	8.39 (11) 6.19 (11)	3.02 (15) 2.23 (15)	3.02 (16) 2.23 (16)	4.56 (16) 3.68 (12)	1.64 (16) 1.32 (16)	1.64 (17) 1.32 (17)	1.08 (12) 7.91 (11)	3.90 (15) 2.85 (15)	3.90 (16) 2.85 (16)
25	Structural Support Supply Plenum Press.	7.0 19.5	-30.8 -28.8	2.03 (-2) 1.49 (-2)	6.48 (6) 4.73 (6)	6.48 (6) 4.73 (6)	6.48 (7) 4.73 (7)	8.39 (11) 6.19 (11)	3.02 (15) 2.23 (15)	3.02 (16) 2.23 (16)	4.56 (16) 3.68 (12)	1.64 (16) 1.32 (16)	1.64 (17) 1.32 (17)	1.08 (12) 7.91 (11)	3.90 (15) 2.85 (15)	3.90 (16) 2.85 (16)
26	NSS Inlet From Turbine Pressure	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)

*Consistent with designation in Table 1 of EC-677565.

**Per Reference *3.

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91 (-3) = 8.91×10^{-3}
2) Data presented for 1511 MW thermal power.

TABLE 11-4
(Continued)RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR GRAPHITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E<0.4 eV			0.4 eV<E<1.0 MeV			E>1.0 MeV		
		Radius (In)	Station (In)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
27	Reflector Inlet/ Turbine Inlet Axial Differential Pressure	7.0 19.5	-30.8 -28.8	2.03 (-2) 1.49 (-2)	6.48 (6) 4.73 (6)	6.48 (6) 4.73 (6)	6.48 (7) 4.73 (7)	8.39 (11) 6.19 (11)	3.02 (15) 2.23 (15)	3.02 (16) 2.23 (16)	4.56 (12) 3.68 (12)	1.64 (16) 1.32 (16)	1.64 (17) 1.32 (17)	1.08 (12) 7.91 (11)	3.90 (15) 2.85 (15)	3.90 (16) 2.85 (16)
28	Reflector Inlet/Core Outlet Differential Pressure	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)
29	Core Axial Differential Pressure	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)
30	NSS Inlet From Turbine to Core Inlet Differential Pressure	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)
31	Structural Support System Differential Pressure (SSCA Inlet to Reflector Outlet)	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)
32	Seal Chamber to Core Exit Differential Pressure	0.0 19.4	-24.5 -24.5	2.85 (-2) 2.33 (-2)	9.05 (6) 7.38 (6)	9.05 (6) 7.38 (6)	9.05 (7) 7.38 (7)	7.20 (11) 4.73 (11)	2.59 (15) 1.70 (15)	2.59 (16) 1.70 (16)	8.04 (12) 5.43 (12)	2.89 (16) 1.95 (16)	2.89 (17) 1.95 (17)	2.06 (12) 1.15 (12)	7.43 (15) 4.14 (15)	7.43 (16) 4.14 (16)
33	Core Nuclear Power (Low Range)	28.08 23.0	26.0 - 4.0	3.40 (0) 3.40 (0)	1.15 (9) 1.11 (9)	1.15 (9) 1.11 (9)	1.15 (10) 1.11 (10)	8.67 (13) 1.31 (14)	3.12 (17) 4.71 (17)	3.12 (18) 4.71 (18)	4.17 (14) 8.26 (14)	1.50 (18) 2.98 (18)	1.50 (19) 2.98 (19)	5.30 (13) 1.01 (14)	1.91 (17) 3.64 (17)	1.91 (18) 3.64 (18)
34	Core Nuclear Power (Intermediate Range)	28.08 23.0	26.0 - 4.0	3.40 (0) 3.40 (0)	1.15 (9) 1.11 (9)	1.15 (9) 1.11 (9)	1.15 (10) 1.11 (10)	8.67 (13) 1.31 (14)	3.12 (17) 4.71 (17)	3.12 (18) 4.71 (18)	4.17 (14) 8.26 (14)	1.50 (18) 2.98 (18)	1.50 (19) 2.98 (19)	5.30 (13) 1.01 (14)	1.91 (17) 3.64 (17)	1.91 (18) 3.64 (18)
35	Core Nuclear Power (High Range)	28.08 23.0	26.0 - 4.0	3.40 (0) 3.40 (0)	1.15 (9) 1.11 (9)	1.15 (9) 1.11 (9)	1.15 (10) 1.11 (10)	8.67 (13) 1.31 (14)	3.12 (17) 4.71 (17)	3.12 (18) 4.71 (18)	4.17 (14) 8.26 (14)	1.50 (18) 2.98 (18)	1.50 (19) 2.98 (19)	5.30 (13) 1.01 (14)	1.91 (17) 3.64 (17)	1.91 (18) 3.64 (18)

*Consistent with designation in Table 1 of EC-677565.

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91 (-3) = 8.91×10^{-3}
2) Data presented for 1511 MW thermal power.

TABLE 11-4
(Continued)

RADIATION ENVIRONMENT AT INSTRUMENTATION LOCATIONS
FOR GRAPHITE FUEL CORE (DWG. 939J723)

Item*	Measurement*	Position		Gamma Environment				Neutron Environment								
								E<0.4 eV			0.4 eV<E<1.0 MeV			E>1.0 MeV		
		Radius (in)	Station (in)	Heating Rate	Dose Rate	1 Hour Dose	10 Hour Dose	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence	Flux	1 Hour Fluence	10 Hour Fluence
				(w/gm(SS))	(Rad-C/hr)	(Rad-C)	(Rad-C)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)	(n/cm ² -sec)	(n/cm ²)	(n/cm ²)
36	Gamma Dose Rate	0.0	-24.5	2.85 (-2)	9.05 (6)	9.05 (6)	9.05 (7)	7.20 (11)	2.59 (15)	2.59 (16)	8.04 (12)	2.89 (16)	2.89 (17)	2.06 (12)	7.43 (15)	7.43 (16)
		19.4	-24.5	2.33 (-2)	7.38 (6)	7.38 (6)	7.38 (7)	4.73 (11)	1.70 (15)	1.70 (16)	5.43 (12)	1.95 (16)	1.95 (17)	1.15 (12)	4.14 (15)	4.14 (16)
		28.08	26.0	3.40 (0)	1.15 (9)	1.15 (9)	1.15 (10)	8.67 (13)	3.12 (17)	3.12 (18)	4.17 (14)	1.50 (18)	1.50 (19)	5.30 (13)	1.91 (17)	1.91 (18)
37	SSCV Position	49.0	-103.0	6.51 (-3)	1.80 (6)	1.80 (6)	1.80 (7)	4.52 (10)	1.63 (14)	1.63 (15)	1.35 (11)	4.86 (14)	4.86 (15)	2.79 (10)	1.00 (14)	1.00 (15)
		42.5	-111.05	3.72 (-3)	1.03 (6)	1.03 (6)	1.03 (7)	3.57 (10)	1.29 (14)	1.29 (15)	1.84 (11)	6.62 (14)	6.62 (15)	2.88 (10)	1.04 (14)	1.04 (15)
38	Control Drum Actuator Shaft Position	24.5	-37.0	9.27 (-3)	2.95 (6)	2.95 (6)	2.95 (7)	1.06 (11)	3.82 (14)	3.82 (15)	6.12 (11)	2.20 (15)	2.20 (16)	2.88 (11)	1.04 (15)	1.04 (16)
39***	Fast Neutron Flux	0.0	-24.5	2.85 (-2)	9.05 (6)	9.05 (6)	9.05 (7)	7.20 (11)	2.59 (15)	2.59 (16)	8.04 (12)	2.89 (16)	2.89 (17)	2.06 (12)	7.43 (15)	7.43 (16)
		19.4	-24.5	2.33 (-2)	7.38 (6)	7.38 (6)	7.38 (7)	4.73 (11)	1.70 (15)	1.70 (16)	5.43 (12)	1.95 (16)	1.95 (17)	1.15 (12)	4.14 (15)	4.14 (16)

*Consistent with designation in Table 1 of EC-677565

***Additional Measurement

NOTE: 1) Numbers in parentheses () refer to powers of ten, i.e., 8.91 (-3) = 8.91×10^{-3}
2) Data presented for 1511 MW thermal power.

TABLE 11-5

TIME INTEGRATED RADIATION ENVIRONMENT AND UNCERTAINTY FACTORS
AT INSTRUMENTATION LOCATIONS FOR THE GRAPHITE FUEL CORE (DWG. 939J725)

LOCATION - STRUCTURAL SUPPORT SYSTEM LOAD
RADIUS (IN) 0.0 STATION (IN) -8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.10E 02	1.07	3.92E 03	1.07	3.94E 03	1.07	4.12E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.97E 15	1.59	1.37E 17	1.59	1.38E 17	1.59	1.44E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.26E 16	1.59	1.13E 18	1.59	1.13E 18	1.59	1.19E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	3.67E 15	1.73	1.27E 17	1.73	1.27E 17	1.73	1.33E 17	1.73
GAMMA RAY DOSE, RADS (CARBON)	9.85E 06	1.07	3.54E 08	1.07	3.56E 08	1.07	3.72E 08	1.07

LOCATION - CONTROL DRUM ACTUATOR SHAFT POSITION
RADIUS (IN) 24.50 STATION (IN) -37.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.53E-01	1.40	5.37E 00	1.40	5.40E 00	1.40	5.64E 00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.75E 12	3.10	1.64E 14	3.10	1.65E 14	3.10	1.73E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.01E 13	3.10	3.49E 14	3.10	3.51E 14	3.10	3.67E 14	3.10
NEUTRONS/CM2, E LT 0.4EV	1.75E 12	3.10	6.05E 13	3.10	6.08E 13	3.10	6.35E 13	3.10
GAMMA RAY DOSE, RADS (CARBON)	1.33E 04	1.40	4.68E 05	1.40	4.71E 05	1.40	4.92E 05	1.40

11-69

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT SYSTEM LOAD
 RADIUS (IN) 15.75 STATION (IN) -8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.67E 01	1.07	2.38E 03	1.07	2.39E 03	1.07	2.50E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.39E 15	1.59	8.26E 16	1.59	8.30E 16	1.59	8.67E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.94E 16	1.59	6.70E 17	1.59	6.73E 17	1.59	7.04E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	2.31E 15	1.73	7.99E 16	1.73	8.03E 16	1.73	8.39E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.96E 06	1.07	2.14E 08	1.07	2.15E 08	1.07	2.25E 08	1.07

LOCATION - NUCLEAR SUBSYSTEM VIBRATION
 RADIUS (IN) 0.0 STATION (IN) -9.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.41E 01	1.07	2.98E 03	1.07	3.00E 03	1.07	3.14E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.78E 15	1.59	9.63E 16	1.59	9.68E 16	1.59	1.01E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.31E 16	1.59	8.00E 17	1.59	8.04E 17	1.59	8.41E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	2.54E 15	1.73	8.78E 16	1.73	8.83E 16	1.73	9.23E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	7.52E 06	1.07	2.68E 08	1.07	2.69E 08	1.07	2.82E 08	1.07

TABLE 11-5 Continued)

LOCATION - NUCLEAR SUBSYSTEM VIBRATION
 RADIUS (IN) 22.00 STATION (IN) -7.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.15E 01	1.07	1.48E 03	1.07	1.49E 03	1.07	1.55E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.20E 15	1.59	4.15E 16	1.59	4.17E 16	1.59	4.35E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.96E 15	1.59	3.45E 17	1.59	3.46E 17	1.59	3.62E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	1.71E 15	1.73	5.92E 16	1.73	5.95E 16	1.73	6.21E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.67E 06	1.07	1.32E 08	1.07	1.33E 08	1.07	1.39E 08	1.07

LOCATION - CORE NUCLEAR POWER(LOW RANGE)
 RADIUS (IN) 28.08 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.97E 01	1.40	1.88E 03	1.40	1.89E 03	1.40	1.98E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.74E 14	3.10	3.02E 16	3.10	3.04E 16	3.10	3.18E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.87E 15	3.10	2.78E 17	3.10	2.39E 17	3.10	2.50E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.95E 16	3.10	4.97E 16	3.10	5.20E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.62E 06	1.40	1.76E 08	1.40	1.77E 08	1.40	1.86E 08	1.40

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TABLE 11-5 (Continued)

LOCATION - CORE NUCLEAR POWER(LOW RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR
WATT-SEC/GM-304 STAINLESS STEEL	5.30E	01	1.12	1.92E	03	1.12	1.93E	03	1.12	2.02E	03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.67E	15	1.13	5.76E	16	1.13	5.79E	16	1.13	6.05E	16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E	16	1.13	4.71E	17	1.13	4.74E	17	1.13	4.95E	17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.16E	15	1.29	7.47E	16	1.29	7.50E	16	1.29	7.84E	16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.77E	06	1.12	1.74E	08	1.12	1.75E	08	1.12	1.83E	08	1.12

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR	NOMINAL	UNC	FACTR
WATT-SEC/GM-304 STAINLESS STEEL	4.66E	01	1.07	1.63E	01	1.07	1.64E	01	1.07	1.72E	01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E	13	1.59	1.18E	15	1.59	1.18E	15	1.59	1.24E	15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E	14	1.59	4.58E	15	1.59	4.61E	15	1.59	4.82E	15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E	13	1.73	4.10E	14	1.73	4.12E	14	1.73	4.31E	14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E	04	1.07	1.44E	06	1.07	1.45E	06	1.07	1.51E	06	1.07

TABLE 11-5 (Continued)

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.89E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

LOCATION - CLUSTER EXIT GAS TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	7.75E 01	1.40	2.97E 03	1.40	2.98E 03	1.40	3.13E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	5.07E 15	3.10	1.75E 17	3.10	1.76E 17	3.10	1.84E 17	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.68E 16	3.10	5.82E 17	3.10	5.85E 17	3.10	6.12E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	8.11E 13	3.10	2.81E 15	3.10	2.82E 15	3.10	2.95E 15	3.10
GAMMA RAY DOSE, RADS (CARBON)	7.40E 06	1.40	2.85E 08	1.40	2.86E 08	1.40	3.00E 08	1.40

TABLE 11-5 (Continued)

LOCATION - CLUSTER EXIT GAS TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.86E 01	1.40	1.85E 03	1.40	1.86E 03	1.40	1.95E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	3.34E 15	3.10	1.16E 17	3.10	1.16E 17	3.10	1.21E 17	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.16E 16	3.10	4.02E 17	3.10	4.04E 17	3.10	4.23E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.94E 14	3.10	6.71E 15	3.10	6.74E 15	3.10	7.04E 15	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.61E 08	1.40	1.77E 08	1.40	1.78E 08	1.40	1.86E 08	1.40

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.91E 01	1.07	2.09E 03	1.07	2.10E 03	1.07	2.19E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.87E 15	1.59	6.47E 16	1.59	6.50E 16	1.59	6.80E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.47E 16	1.59	5.08E 17	1.59	5.11E 17	1.59	5.34E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	6.21E 14	1.73	2.15E 16	1.73	2.16E 16	1.73	2.26E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.43E 06	1.07	1.92E 08	1.07	1.93E 08	1.07	2.02E 08	1.07

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE									
RADIUS (IN)		0.0		STATION (IN)		-17.50			

STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)			
NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR			
WATT-SEC/GM-304 STAINLESS STEEL									
1.23E 01		1.07		4.33E 02		1.07		4.35E 02	
1.07		4.33E 02		1.07		4.35E 02		1.07	
NEUTRONS/CM2, E GT 1.0MEV									
2.80E 14		1.59		9.70E 15		1.59		9.75E 15	
1.59		9.70E 15		1.59		9.75E 15		1.59	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV									
1.27E 15		1.59		4.41E 16		1.59		4.43E 16	
1.59		4.41E 16		1.59		4.43E 16		1.59	
NEUTRONS/CM2, E LT 0.4EV									
1.90E 12		1.73		6.58E 13		1.73		6.61E 13	
1.73		6.58E 13		1.73		6.61E 13		1.73	
GAMMA RAY DOSE, RADS (CARBON)									
1.13E 06		1.07		4.00E 07		1.07		4.02E 07	
1.07		4.00E 07		1.07		4.02E 07		1.07	

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE									
RADIUS (IN)			0.0		STATION (IN)			-22.93	

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)									
NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR									
WATT-SEC/GM-304 STAINLESS STEEL									
NEUTRONS/CM2, E GT 1.0MEV									
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV									
NEUTRONS/CM2, E LT 0.4EV									
GAMMA RAY DOSE, RADS (CARBON)									

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -24.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.72E-01	1.07	1.65E 01	1.07	1.66E 01	1.07	1.74E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.48E 13	1.59	1.20E 15	1.59	1.21E 15	1.59	1.26E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.35E 14	1.59	4.66E 15	1.59	4.69E 15	1.59	4.90E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.13E 13	1.73	3.93E 14	1.73	3.95E 14	1.73	4.12E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.14E 04	1.07	1.46E 06	1.07	1.46E 06	1.07	1.53E 06	1.07

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.91E 01	1.07	1.73E 03	1.07	1.74E 03	1.07	1.82E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.56E 15	1.59	5.41E 16	1.59	5.44E 16	1.59	5.69E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.22E 16	1.59	4.21E 17	1.59	4.23E 17	1.59	4.42E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	5.15E 14	1.73	1.78E 16	1.73	1.79E 16	1.73	1.87E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.50E 06	1.07	1.59E 08	1.07	1.60E 08	1.07	1.67E 08	1.07

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -17.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.03E 01	1.07	3.62E 02	1.07	3.63E 02	1.07	3.80E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.35E 14	1.59	8.12E 15	1.59	8.10E 15	1.59	8.53E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.06E 15	1.59	3.67E 16	1.59	3.69E 16	1.59	3.85E 16	1.59
NEUTRONS/CM2, L LT 0.4EV	1.58E 12	1.73	5.46E 13	1.73	5.49E 13	1.73	5.73E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	9.47E 05	1.07	3.34E 07	1.07	3.36E 07	1.07	3.51E 07	1.07

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -22.93

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.08E 00	1.07	7.30E 01	1.07	7.34E 01	1.07	7.67E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	4.61E 13	1.59	1.59E 15	1.59	1.60E 15	1.59	1.67E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.69E 14	1.59	5.85E 15	1.59	5.88E 15	1.59	6.14E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	2.75E 12	1.73	9.51E 13	1.73	9.56E 13	1.73	9.99E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.86E 05	1.07	6.55E 06	1.07	6.58E 06	1.07	6.88E 06	1.07

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 10.00 STATION (IN) -24.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.04E-01	1.07	1.42E 01	1.07	1.43E 01	1.07	1.49E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.04E 13	1.59	1.05E 15	1.59	1.06E 15	1.59	1.11E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.17E 14	1.59	4.04E 15	1.59	4.07E 15	1.59	4.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.04E 13	1.73	3.59E 14	1.73	3.61E 14	1.73	3.77E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.55E 04	1.07	1.25E 06	1.07	1.26E 06	1.07	1.31E 06	1.07

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -11.83

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.57E 01	1.07	9.08E 02	1.07	9.12E 02	1.07	9.54E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	8.21E 14	1.59	2.84E 16	1.59	2.85E 16	1.59	2.98E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.15E 15	1.59	2.13E 17	1.59	2.14E 17	1.59	2.24E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	3.21E 14	1.73	1.11E 16	1.73	1.12E 16	1.73	1.17E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.34E 06	1.07	8.30E 07	1.07	8.34E 07	1.07	8.72E 07	1.07

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -17.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.71E 00	1.07	2.01E 02	1.07	2.02E 02	1.07	2.11E 02	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.38E 14	1.59	4.78E 15	1.59	4.81E 15	1.59	5.03E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.14E 14	1.59	2.47E 16	1.59	2.48E 16	1.59	2.60E 16	1.59
NEUTRONS/CM2, E LT 0.4EV	2.21E 12	1.73	7.64E 13	1.73	7.68E 13	1.73	8.03E 13	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.22E 05	1.07	1.84E 07	1.07	1.85E 07	1.07	1.93E 07	1.07

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -22.93

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.26E 00	1.07	4.42E 01	1.07	4.44E 01	1.07	4.64E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.85E 13	1.59	9.85E 14	1.59	9.90E 14	1.59	1.03E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.28E 14	1.59	4.42E 15	1.59	4.44E 15	1.59	4.64E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	2.96E 12	1.73	1.02E 14	1.73	1.03E 14	1.73	1.08E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.12E 05	1.07	3.93E 06	1.07	3.94E 06	1.07	4.12E 06	1.07

TABLE 11-5 (Continued)

LOCATION - CENTRAL SHIELD MATERIAL TEMPERATURE
 RADIUS (IN) 19.40 STATION (IN) -24.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.98E-01	1.07	1.39E 01	1.07	1.40E 01	1.07	1.46E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.96E 13	1.59	6.79E 14	1.59	6.83E 14	1.59	7.14E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.24E 13	1.59	3.20E 15	1.59	3.21E 15	1.59	3.36E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.46E 12	1.73	2.58E 14	1.73	2.59E 14	1.73	2.71E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.48E 04	1.07	1.22E 06	1.07	1.23E 06	1.07	1.28E 06	1.07

LOCATION - CORE STATION 0.2 FUEL ELEMENT TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 0.23

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.23E 02	1.12	1.62E 04	1.12	1.63E 04	1.12	1.71E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.90E 16	1.02	6.58E 17	1.02	6.62E 17	1.02	6.91E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	9.96E 16	1.02	3.45E 18	1.02	3.46E 18	1.02	3.62E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	3.94E 14	1.29	1.36E 16	1.29	1.37E 16	1.29	1.43E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.09E 07	1.12	1.58E 09	1.12	1.59E 09	1.12	1.66E 09	1.12

TABLE 11-5 (Continued)

LOCATION - CORE STATION 0.2 FUEL ELEMENT TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 0.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.71E 02	1.12	1.04E 04	1.12	1.05E 04	1.12	1.10E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.38E 16	1.02	4.79E 17	1.02	4.82E 17	1.02	5.03E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.05E 16	1.02	2.09E 18	1.02	2.10E 18	1.02	2.20E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	2.47E 14	1.29	8.56E 15	1.29	8.60E 15	1.29	8.99E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	2.62E 07	1.12	1.02E 09	1.12	1.02E 09	1.12	1.07E 09	1.12

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.29E 03	1.12	5.14E 04	1.12	5.17E 04	1.12	5.43E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	7.42E 16	1.02	2.57E 18	1.02	2.58E 18	1.02	2.70E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.78E 17	1.02	1.31E 19	1.02	1.31E 19	1.02	1.37E 19	1.02
NEUTRONS/CM2, E LT 0.4EV	3.00E 14	1.29	1.04E 16	1.29	1.04E 16	1.29	1.09E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.29E 08	1.12	5.12E 09	1.12	5.15E 09	1.12	5.40E 09	1.12

TABLE 11-5 (Continued)

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE									
RADIUS (IN)		0.0		STATION (IN)		18.00			

STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)			
NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR			
WATT-SEC/GM-304 STAINLESS STEEL	1.25E 03	1.12	4.98E 04	1.12	5.01E 04	1.12	5.26E 04	1.12	
NEUTRONS/CM2, E GT 1.0MEV	7.27E 16	1.02	2.51E 18	1.02	2.53E 18	1.02	2.64E 18	1.02	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.68E 17	1.02	1.27E 19	1.02	1.28E 19	1.02	1.34E 19	1.02	
NEUTRONS/CM2, E LT 0.4EV	2.90E 14	1.29	1.00E 16	1.29	1.01E 16	1.29	1.05E 16	1.29	
GAMMA RAY DOSE, RADS (CARBON)	1.25E 08	1.12	4.96E 09	1.12	4.99E 09	1.12	5.23E 09	1.12	

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE									
RADIUS (IN)		15.75		STATION (IN)		26.00			

STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)			
NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR		NOMINAL UNC FACTR			
WATT-SEC/GM-304 STAINLESS STEEL									
8.69E 02		1.12		3.46E 04		1.12		3.48E 04	
1.12		3.65E 04		1.12		3.48E 04		1.12	
NEUTRONS/CM2, E GT 1.0MEV									
5.52E 16		1.02		1.91E 18		1.02		1.92E 18	
1.02		2.01E 18		1.02		1.92E 18		1.02	
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV									
2.30E 17		1.02		7.97E 18		1.02		8.01E 18	
1.02		8.37E 18		1.02		8.01E 18		1.02	
NEUTRONS/CM2, E LT 0.4EV									
2.09E 14		1.29		7.23E 15		1.29		7.27E 15	
1.29		7.65E 15		1.29		7.27E 15		1.29	
GAMMA RAY DOSE, RADS (CARBON)									
8.64E 07		1.12		3.45E 09		1.12		3.46E 09	
1.12		3.64E 09		1.12		3.46E 09		1.12	

TABLE 11-5 (Continued)

LOCATION - CORE UNFUELED ELEMENT TEMPERATURE
 RADIUS (IN) 15.75 STATION (IN) 19.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.43E 02	1.12	3.36E 04	1.12	3.38E 04	1.12	3.54E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	5.41E 16	1.02	1.87E 18	1.02	1.88E 18	1.02	1.97E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.25E 17	1.02	7.77E 18	1.02	7.81E 18	1.02	8.16E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	2.02E 14	1.29	6.98E 15	1.29	7.02E 15	1.29	7.33E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	8.37E 07	1.12	3.34E 09	1.12	3.36E 09	1.12	3.53E 09	1.12

LOCATION - NSS-OUTLET-TO-TURBINE GAS TEMPERATURE
 RADIUS (IN) 7.30 STATION (IN) -35.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.73E-01	1.40	9.56E 00	1.40	9.61E 00	1.40	1.00E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.27E 13	3.10	4.38E 14	3.10	4.40E 14	3.10	4.60E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.82E 13	3.10	1.67E 15	3.10	1.68E 15	3.10	1.75E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.01E 13	3.10	3.48E 14	3.10	3.50E 14	3.10	3.65E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.41E 04	1.40	8.46E 05	1.40	8.50E 05	1.40	8.88E 05	1.40

TABLE 11-5 (Continued)

LOCATION - NSS INLET FROM TURBINE GAS TEMPERATURE
 RADIUS (IN) 3.00 STATION (IN) -32.50

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	3.23E-01	1.40	1.12E 01	1.40	1.12E 01	1.40	1.18E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.55E 13	3.10	5.36E 14	3.10	5.39E 14	3.10	5.63E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.49E 13	3.10	2.25E 15	3.10	2.26E 15	3.10	2.36E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.42E 13	3.10	4.93E 14	3.10	4.95E 14	3.10	5.17E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.82E 04	1.40	9.92E 05	1.40	9.97E 05	1.40	1.04E 06	1.40

LOCATION - FINAL SEAL EXIT GAS TEMPERATURE
 RADIUS (IN) 19.50 STATION (IN) 49.00

	STATE POINT (B)		STATE POINT (C)		STATE POINT (D)		STATE POINT (E)	
	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR	NOMINAL	UNC FACTR
WATT-SEC/GM-304 STAINLESS STEEL	1.19E 02	1.12	4.65E 03	1.12	4.67E 03	1.12	4.90E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	7.10E 15	1.13	2.46E 17	1.13	2.47E 17	1.13	2.58E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.41E 16	1.13	1.53E 18	1.13	1.54E 18	1.13	1.60E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	1.70E 15	1.29	5.87E 16	1.29	5.90E 16	1.29	6.17E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.16E 07	1.12	4.55E 08	1.12	4.58E 08	1.12	4.80E 08	1.12

TABLE 11-5 (Continued)

LOCATION - CORE INLET PLENUM TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -3.0

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.64E 02	1.07	9.65E 03	1.07	9.70E 03	1.07	1.01E 04	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.08E 16	1.59	3.73E 17	1.59	3.75E 17	1.59	3.92E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.69E 16	1.59	2.31E 18	1.59	2.33E 18	1.59	2.43E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	1.69E 15	1.73	5.83E 16	1.73	5.86E 16	1.73	6.13E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.45E 07	1.07	9.03E 08	1.07	9.08E 08	1.07	9.50E 08	1.07

LOCATION - CORE INLET PLENUM TEMPERATURE
 RADIUS (IN) 19.00 STATION (IN) -3.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.09E 02	1.07	3.98E 03	1.07	4.03E 03	1.07	4.19E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	4.67E 15	1.59	1.62E 17	1.59	1.64E 17	1.59	1.70E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.03E 16	1.59	1.05E 18	1.59	1.05E 18	1.59	1.10E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	1.76E 15	1.73	6.09E 16	1.73	6.11E 16	1.73	6.39E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.00E 07	1.07	3.68E 08	1.07	3.70E 08	1.07	3.88E 08	1.07

TABLE 11-5 (Continued)

LOCATION - REFLECTOR INLET PLENUM TEMPERATURE
 RADIUS (IN) 24.50 STATION (IN) 55.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.39E 01	1.40	8.97E 02	1.40	9.02E 02	1.40	9.44E 02	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.02E 14	3.10	2.78E 16	3.10	2.79E 16	3.10	2.92E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.39E 15	3.10	1.07E 17	3.10	1.88E 17	3.10	1.96E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.13E 15	3.10	3.92E 16	3.10	3.94E 16	3.10	4.12E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.23E 06	1.40	8.42E 07	1.40	8.47E 07	1.40	8.87E 07	1.40

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.82E 01	1.12	1.80E 03	1.12	1.81E 03	1.12	1.90E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	9.70E 14	1.13	3.36E 16	1.13	3.37E 16	1.13	3.53E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.10E 16	1.13	3.80E 17	1.13	3.82E 17	1.13	3.99E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.23E 15	1.29	7.73E 16	1.29	7.77E 16	1.29	8.12E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.45E 06	1.12	1.68E 08	1.12	1.69E 08	1.12	1.77E 08	1.12

TABLE 11-5 (Continued)

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	0.77E 01	1.12	3.31E 03	1.12	3.33E 03	1.12	3.49E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.90E 15	1.13	6.57E 16	1.13	6.60E 16	1.13	6.90E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.24E 16	1.13	7.77E 17	1.13	7.81E 17	1.13	8.16E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	4.63E 15	1.29	1.60E 17	1.29	1.61E 17	1.29	1.68E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	8.14E 06	1.12	3.10E 08	1.12	3.12E 08	1.12	3.27E 08	1.12

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 26.50 STATION (IN) 48.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.30E 01	1.12	1.24E 03	1.12	1.25E 03	1.12	1.31E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	6.57E 14	1.13	2.27E 16	1.13	2.29E 16	1.13	2.39E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.37E 15	1.13	2.55E 17	1.13	2.56E 17	1.13	2.68E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	1.49E 15	1.29	5.17E 16	1.29	5.20E 16	1.29	5.43E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.06E 06	1.12	1.16E 08	1.12	1.17E 08	1.12	1.22E 08	1.12

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TABLE 11-5 (Continued)

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 27.20 STATION (IN) 4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.10E 02	1.12	4.21E 03	1.12	4.23E 03	1.12	4.44E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	5.78E 15	1.13	2.00E 17	1.13	2.01E 17	1.13	2.10E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.90E 16	1.13	2.04E 18	1.13	2.05E 18	1.13	2.14E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	4.41E 15	1.29	1.53E 17	1.29	1.53E 17	1.29	1.60E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.05E 07	1.12	4.02E 08	1.12	4.04E 08	1.12	4.24E 08	1.12

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 22.20 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.08E 02	1.12	8.01E 03	1.12	8.05E 03	1.12	8.44E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.13E 16	1.13	3.92E 17	1.13	3.94E 17	1.13	4.12E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.21E 17	1.13	4.19E 18	1.13	4.21E 18	1.13	4.40E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	8.98E 15	1.29	3.11E 17	1.29	3.12E 17	1.29	3.26E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.99E 07	1.12	7.70E 08	1.12	7.75E 08	1.12	8.12E 08	1.12

TABLE 11-5 (Continued)

LOCATION - REFLECTOR MATERIAL TEMPERATURE
 RADIUS (IN) 22.20 STATION (IN) 48.0

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	7.40E 01	1.12	2.84E 03	1.12	2.85E 03	1.12	2.99E 03	1.12
NEUTRONS/CM2, E LT 1.0MEV	3.83E 15	1.13	1.32E 17	1.13	1.33E 17	1.13	1.39E 17	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.89E 16	1.13	1.35E 18	1.13	1.35E 18	1.13	1.42E 18	1.13
NEUTRONS/CM2, E LT 0.4EV	2.88E 15	1.29	9.97E 16	1.29	1.00E 17	1.29	1.05E 17	1.29
GAMMA RAY DOSE, RADS (CARBON)	7.06E 06	1.12	2.72E 08	1.12	2.74E 08	1.12	2.87E 08	1.12

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM GAS TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -5.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.71E 02	1.07	6.22E 03	1.07	6.23E 03	1.07	6.52E 03	1.07
NEUTRONS/CM2, E LT 1.0MEV	6.99E 15	1.59	2.42E 17	1.59	2.43E 17	1.59	2.54E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.16E 16	1.59	1.78E 18	1.59	1.79E 18	1.59	1.87E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	3.86E 15	1.73	1.34E 17	1.73	1.34E 17	1.73	1.40E 17	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.56E 07	1.07	5.72E 08	1.07	5.75E 08	1.07	6.01E 08	1.07

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM GAS TEMPERATURE
 RADIUS (IN) 17.00 STATION (IN) -5.20

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	9.27E 01	1.07	3.35E 03	1.07	3.37E 03	1.07	3.52E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.82E 15	1.59	1.32E 17	1.59	1.33E 17	1.59	1.39E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.77E 16	1.59	9.59E 17	1.59	9.64E 17	1.59	1.01E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	2.40E 15	1.73	8.31E 16	1.73	8.35E 16	1.73	8.73E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	8.43E 06	1.07	3.06E 08	1.07	3.08E 08	1.07	3.22E 08	1.07

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.83E 02	1.12	3.50E 04	1.12	3.52E 04	1.12	3.69E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	5.24E 16	1.02	1.81E 18	1.02	1.82E 18	1.02	1.91E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.59E 17	1.02	8.97E 18	1.02	9.01E 18	1.02	9.42E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	2.07E 14	1.29	7.15E 15	1.29	7.18E 15	1.29	7.51E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	8.77E 07	1.12	3.48E 09	1.12	3.50E 09	1.12	3.67E 09	1.12

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.29E 03	1.12	5.14E 04	1.12	5.17E 04	1.12	5.43E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	7.42E 16	1.02	2.57E 18	1.02	2.56E 18	1.02	2.70E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.78E 17	1.02	1.31E 19	1.02	1.31E 19	1.02	1.37E 19	1.02
NEUTRONS/CM2, E LT 0.4EV	3.00E 14	1.29	1.04E 16	1.29	1.04E 16	1.29	1.09E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	1.29E 08	1.12	5.12E 09	1.12	5.15E 09	1.12	5.40E 09	1.12

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) 50.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.15E 02	1.12	1.24E 04	1.12	1.25E 04	1.12	1.31E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.74E 16	1.02	6.02E 17	1.02	6.05E 17	1.02	6.32E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.33E 16	1.02	2.88E 18	1.02	2.9CE 18	1.02	3.03E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	6.71E 13	1.29	2.32E 15	1.29	2.33E 15	1.29	2.44E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	3.11E 07	1.12	1.23E 09	1.12	1.23E 09	1.12	1.30E 09	1.12

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 8.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.83E 02	1.12	2.32E 04	1.12	2.33E 04	1.12	2.45E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	3.84E 16	1.02	1.33E 18	1.02	1.33E 18	1.02	1.40E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.56E 17	1.02	5.38E 18	1.02	5.41E 18	1.02	5.65E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	1.63E 14	1.29	5.64E 15	1.29	5.67E 15	1.29	5.92E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	5.79E 07	1.12	2.31E 09	1.12	2.32E 09	1.12	2.43E 09	1.12

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	8.55E 02	1.12	3.41E 04	1.12	3.42E 04	1.12	3.59E 04	1.12
NEUTRONS/CM2, E GT 1.0MEV	5.41E 16	1.02	1.87E 18	1.02	1.88E 18	1.02	1.97E 18	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.26E 17	1.02	7.81E 18	1.02	7.85E 18	1.02	8.21E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	2.42E 14	1.29	8.38E 15	1.29	8.42E 15	1.29	8.81E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	8.49E 07	1.12	3.39E 09	1.12	3.41E 09	1.12	3.58E 09	1.12

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT LINER TEMPERATURE
 RADIUS (IN) 16.00 STATION (IN) 50.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.04E 02	1.12	8.06E 03	1.12	8.10E 03	1.12	8.50E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.26E 16	1.02	4.36E 17	1.02	4.39E 17	1.02	4.59E 17	1.02
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	5.01E 16	1.02	1.73E 18	1.02	1.74E 18	1.02	1.82E 18	1.02
NEUTRONS/CM2, E LT 0.4EV	5.70E 13	1.29	1.97E 15	1.29	1.98E 15	1.29	2.07E 15	1.29
GAMMA RAY DOSE, RADS (CARBON)	2.01E 07	1.12	7.98E 08	1.12	8.02E 08	1.12	8.41E 08	1.12

LOCATION - STRUCTURAL SUPPORT EXHAUST PLENUM TEMPERATURE
 RADIUS (IN) 21.50 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.78E 01	1.07	1.71E 03	1.07	1.72E 03	1.07	1.80E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.49E 15	1.59	5.17E 16	1.59	5.19E 16	1.59	5.43E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.21E 16	1.59	4.17E 17	1.59	4.20E 17	1.59	4.38E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	1.88E 15	1.73	6.51E 16	1.73	6.54E 16	1.73	6.83E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.25E 06	1.07	1.53E 08	1.07	1.54E 08	1.07	1.61E 08	1.07

TABLE 11-5 (Continued)

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 0.0 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.38E 02	1.07	4.98E 03	1.07	5.01E 03	1.07	5.24E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	5.40E 15	1.59	1.87E 17	1.59	1.89E 17	1.59	1.96E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	4.23E 16	1.59	1.46E 18	1.59	1.47E 18	1.59	1.54E 19	1.59
NEUTRONS/CM2, E LT 0.4EV	4.11E 15	1.73	1.42E 17	1.73	1.43E 17	1.73	1.49E 17	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.25E 07	1.07	4.55E 08	1.07	4.57E 08	1.07	4.78E 08	1.07

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 8.00 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.23E 02	1.07	4.42E 03	1.07	4.44E 03	1.07	4.65E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	4.91E 15	1.59	1.70E 17	1.59	1.71E 17	1.59	1.78E 17	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.78E 16	1.59	1.31E 18	1.59	1.31E 18	1.59	1.37E 18	1.59
NEUTRONS/CM2, E LT 0.4EV	3.63E 15	1.73	1.26E 17	1.73	1.26E 17	1.73	1.32E 17	1.73
GAMMA RAY DOSE, RADS (CARBON)	1.11E 07	1.07	4.04E 08	1.07	4.06E 08	1.07	4.24E 08	1.07

TABLE 11-5 (Continued)

LOCATION - SUPPORT PLATE MATERIAL TEMPERATURE
 RADIUS (IN) 19.00 STATION (IN) -6.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.38E 01	1.07	2.29E 03	1.07	2.30E 03	1.07	2.40E 03	1.07
NEUTRONS/CM2, E GT 1.0MEV	2.31E 15	1.59	7.97E 16	1.59	8.03E 16	1.59	8.40E 16	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.82E 16	1.59	6.28E 17	1.59	6.32E 17	1.59	6.60E 17	1.59
NEUTRONS/CM2, E LT 0.4EV	2.22E 15	1.73	7.69E 16	1.73	7.72E 16	1.73	8.07E 16	1.73
GAMMA RAY DOSE, RADS (CARBON)	5.71E 06	1.07	2.06E 08	1.07	2.07E 08	1.07	2.16E 08	1.07

LOCATION - GAMMA DOSE RATE
 RADIUS (IN) 28.08 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.97E 01	1.40	1.88E 03	1.40	1.89E 03	1.40	1.98E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.74E 14	3.10	3.02E 16	3.10	3.04E 16	3.10	3.18E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.87E 15	3.10	2.38E 17	3.10	2.39E 17	3.10	2.50E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.95E 16	3.10	4.97E 16	3.10	5.20E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.62E 06	1.40	1.76E 08	1.40	1.77E 08	1.40	1.86E 08	1.40

TABLE 11-5 (Continued)

LOCATION - REFLECTOR INLET/TURBINE INLET AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.32E-01	1.40	1.16E 01	1.40	1.17E 01	1.40	1.22E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.79E 13	3.10	6.18E 14	3.10	6.21E 14	3.10	6.49E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.52E 13	3.10	2.60E 15	3.10	2.61E 15	3.10	2.73E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.38E 13	3.10	4.78E 14	3.10	4.81E 14	3.10	5.03E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.93E 04	1.40	1.03E 06	1.40	1.04E 06	1.40	1.08E 06	1.40

LOCATION - REFLECTOR INLET/TURBINE INLET AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.45E-01	1.07	8.56E 00	1.07	8.60E 00	1.07	8.99E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.30E 13	1.59	4.51E 14	1.59	4.54E 14	1.59	4.74E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.06E 13	1.59	2.10E 15	1.59	2.11E 15	1.59	2.20E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.02E 13	1.73	3.53E 14	1.73	3.55E 14	1.73	3.71E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.16E 04	1.07	7.54E 05	1.07	7.58E 05	1.07	7.92E 05	1.07

TABLE 11-5 (Continued)

LOCATION - SEAL CHAMBER TO CORE EXIT DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.06E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - SEAL CHAMBER TO CORE EXIT DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.84E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT SYSTEM DIFFERENTIAL PRESSURE(SSCA INLET TO REFL OUT)
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - STRUCTURAL SUPPORT SYSTEM DIFFERENTIAL PRESSURE(SSCA INLET TO REFL OUT)
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.84E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11- 5(Continued)

LOCATION - REFLECTOR/INLET CORE OUTLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - REFLECTOR/INLET CORE OUTLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.89E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11-5 (Continued)

LOCATION - REFLECTOR INLET PLENUM PRESSURE

RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.32E-01	1.40	1.16E 01	1.40	1.17E 01	1.40	1.22E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.79E 13	3.10	6.18E 14	3.10	6.21E 14	3.10	6.49E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.52E 13	3.10	2.6CE 15	3.10	2.61E 15	3.10	2.73E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.38E 13	3.10	4.78E 14	3.10	4.91E 14	3.10	5.03E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.93E 04	1.40	1.03E 06	1.40	1.04E 06	1.40	1.08E 06	1.40

LOCATION - REFLECTOR INLET PLENUM PRESSURE

RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.45E-01	1.07	8.56E 00	1.07	8.6CL 00	1.07	8.99E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.30E 13	1.59	4.51E 14	1.59	4.54E 14	1.59	4.74E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.06E 13	1.59	2.10E 15	1.59	2.11E 15	1.59	2.20E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.02E 13	1.73	3.53E 14	1.73	3.55E 14	1.73	3.71E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.16E 04	1.07	7.54E 05	1.07	7.58E 05	1.07	7.92E 05	1.07

TABLE 11-5 (Continued)

LOCATION - NSS INLET FROM TURBINE PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.50E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - NSS INLET FROM TURBINE PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.89E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

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TABLE 11-5 (Continued)

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.32E-01	1.40	1.16E 01	1.40	1.17E 01	1.40	1.22E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.79E 13	3.10	6.18E 14	3.10	6.21E 14	3.10	6.47E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.52E 13	3.10	2.60E 15	3.10	2.61E 15	3.10	2.73E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.38E 13	3.10	4.78E 14	3.10	4.81E 14	3.10	5.03E 14	3.10
GAMMA RAY DOSE, RADS (CARBON)	2.93E 04	1.40	1.03E 06	1.40	1.04E 06	1.40	1.08E 06	1.40

LOCATION - STRUCTURAL SUPPORT SUPPLY PLENUM PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.45E-01	1.07	8.56E 00	1.07	8.60E 00	1.07	8.99E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.30E 13	1.59	4.51E 14	1.59	4.54E 14	1.59	4.74E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.06E 13	1.59	2.10E 15	1.59	2.11E 15	1.59	2.20E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.02E 13	1.73	3.53E 14	1.73	3.55E 14	1.73	3.71E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	2.16E 04	1.07	7.54E 05	1.07	7.58E 05	1.07	7.92E 05	1.07

TABLE 11-5 (Continued)

LOCATION - CORE AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - CORE AXIAL DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.89E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11-5 (Continued)

LOCATION - NSS OUTLET TO TURBINE PRESSURE
 RADIUS (IN) 7.00 STATION (IN) -30.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.32E-01	1.40	1.16E 01	1.40	1.17E 01	1.40	1.22E 01	1.40
NEUTRONS/CM2, E GT 1.0MEV	1.79E 13	3.10	6.18E 14	3.10	6.21E 14	3.10	6.49E 14	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	7.52E 13	3.10	2.60E 15	3.10	2.61E 15	3.10	2.73E 15	3.10
NEUTRONS/CM2, E LT 0.4EV	1.38E 13	3.10	4.78E 14	3.10	4.81E 14	3.10	5.03E 14	3.10
GAMMA RAY DOSE, RAD(S CARBON)	2.93E 04	1.40	1.03E 06	1.40	1.04E 06	1.40	1.08E 06	1.40

LOCATION - NSS OUTLET TO TURBINE PRESSURE
 RADIUS (IN) 19.50 STATION (IN) -28.80

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	2.45E-01	1.07	8.56E 00	1.07	8.60E 00	1.07	8.99E 00	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.30E 13	1.59	4.51E 14	1.59	4.54E 14	1.59	4.74E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.06E 13	1.59	2.10E 15	1.59	2.11E 15	1.59	2.20E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.02E 13	1.73	3.53E 14	1.73	3.55E 14	1.73	3.71E 14	1.73
GAMMA RAY DOSE, RAD(S CARBON)	2.16E 04	1.07	7.54E 05	1.07	7.58E 05	1.07	7.92E 05	1.07

TABLE 11-5 (Continued)

LOCATION - NSS INLET FROM TURBINE TO CORE INLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - NSS INLET FROM TURBINE TO CORE INLET DIFFERENTIAL PRESSURE
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.89E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.93E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11-5 (Continued)

LOCATION - FAST NEUTRON FLUX
 RADIUS (IN) 0.0 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.66E-01	1.07	1.63E 01	1.07	1.64E 01	1.07	1.72E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	3.40E 13	1.59	1.18E 15	1.59	1.18E 15	1.59	1.24E 15	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.32E 14	1.59	4.58E 15	1.59	4.61E 15	1.59	4.82E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	1.19E 13	1.73	4.10E 14	1.73	4.12E 14	1.73	4.31E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	4.09E 04	1.07	1.44E 06	1.07	1.45E 06	1.07	1.51E 06	1.07

LOCATION - FAST NEUTRON FLUX
 RADIUS (IN) 19.40 STATION (IN) -24.50

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	3.83E-01	1.07	1.34E 01	1.07	1.35E 01	1.07	1.41E 01	1.07
NEUTRONS/CM2, E GT 1.0MEV	1.90E 13	1.59	6.56E 14	1.59	6.60E 14	1.59	6.84E 14	1.59
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	8.95E 13	1.59	3.10E 15	1.59	3.11E 15	1.59	3.25E 15	1.59
NEUTRONS/CM2, E LT 0.4EV	7.79E 12	1.73	2.70E 14	1.73	2.71E 14	1.73	2.83E 14	1.73
GAMMA RAY DOSE, RADS (CARBON)	3.35E 04	1.07	1.18E 06	1.07	1.18E 06	1.07	1.23E 06	1.07

TABLE 11-5 (Continued)

LOCATION - CORE NUCLEAR POWER (INTERMEDIATE RANGE)
 RADIUS (IN) 26.08 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.97E 01	1.40	1.83E 03	1.40	1.87E 03	1.40	1.98E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.74E 14	3.10	3.02E 16	3.10	3.04E 16	3.10	3.14E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.87E 15	3.10	2.33E 17	3.10	2.53E 17	3.10	2.50E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.95E 16	3.10	4.97E 16	3.10	5.20E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.62E 06	1.40	1.76E 08	1.40	1.77E 08	1.40	1.86E 08	1.40

LOCATION - CORE NUCLEAR POWER (INTERMEDIATE RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.30E 01	1.12	1.92E 03	1.12	1.93E 03	1.12	2.02E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.67E 15	1.13	5.76E 16	1.13	5.79E 16	1.13	6.05E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 16	1.13	4.71E 17	1.13	4.74E 17	1.13	4.95E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.16E 15	1.29	7.47E 16	1.29	7.50E 16	1.29	7.84E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.77E 06	1.12	1.74E 08	1.12	1.75E 08	1.12	1.83E 08	1.12

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TABLE 11-5 (Continued)

LOCATION - CORE NUCLEAR POWER(HIGH RANGE)
 RADIUS (IN) 28.00 STATION (IN) 26.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	4.97E 01	1.40	1.88E 03	1.40	1.89E 03	1.40	1.98E 03	1.40
NEUTRONS/CM2, E GT 1.0MEV	8.74E 14	3.10	3.02E 16	3.10	3.04E 16	3.10	3.18E 16	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	6.87E 15	3.10	2.38E 17	3.10	2.39E 17	3.10	2.50E 17	3.10
NEUTRONS/CM2, E LT 0.4EV	1.43E 15	3.10	4.95E 16	3.10	4.97E 16	3.10	5.20E 16	3.10
GAMMA RAY DOSE, RADS (CARBON)	4.62E 06	1.40	1.76E 08	1.40	1.77E 08	1.40	1.86E 08	1.40

LOCATION - CORE NUCLEAR POWER(HIGH RANGE)
 RADIUS (IN) 23.00 STATION (IN) -4.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	5.30E 01	1.12	1.92E 03	1.12	1.93E 03	1.12	2.02E 03	1.12
NEUTRONS/CM2, E GT 1.0MEV	1.67E 15	1.13	5.76E 16	1.13	5.79E 16	1.13	6.05E 16	1.13
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	1.36E 16	1.13	4.71E 17	1.13	4.74E 17	1.13	4.95E 17	1.13
NEUTRONS/CM2, E LT 0.4EV	2.16E 15	1.29	7.47E 16	1.29	7.50E 16	1.29	7.84E 16	1.29
GAMMA RAY DOSE, RADS (CARBON)	4.77E 06	1.12	1.74E 08	1.12	1.75E 08	1.12	1.83E 08	1.12

TABLE 11-5(Continued)

LOCATION - SSCV POSITION
 RADIUS (IN) 49.00 STATION (IN) -103.00

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	1.07E-01	1.40	3.74E-00	1.40	3.74E-00	1.40	3.94E-00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.60E-11	3.10	1.59E-13	3.10	1.60E-13	3.10	1.67E-13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	2.22E-12	3.10	7.70E-13	3.10	7.74E-13	3.10	8.09E-13	3.10
NEUTRONS/CM2, E LT 0.4EV	7.45E-11	3.10	2.58E-13	3.10	2.54E-13	3.10	2.71E-13	3.10
GAMMA RAY DOSE, RAD(S CARBON)	8.25E-03	1.40	2.88E-05	1.40	2.89E-05	1.40	3.02E-05	1.40

LOCATION - SSCV POSITION
 RADIUS (IN) 42.50 STATION (IN) -111.05

STATE POINT (B) STATE POINT (C) STATE POINT (D) STATE POINT (E)

NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR NOMINAL UNC FACTR

WATT-SEC/GM-304 STAINLESS STEEL	6.14E-02	1.40	2.14E-00	1.40	2.15E-00	1.40	2.25E-00	1.40
NEUTRONS/CM2, E GT 1.0MEV	4.75E-11	3.10	1.64E-13	3.10	1.65E-13	3.10	1.73E-13	3.10
NEUTRONS/CM2, 0.4EV LT E LT 1.0MEV	3.03E-12	3.10	1.05E-14	3.10	1.05E-14	3.10	1.10E-14	3.10
NEUTRONS/CM2, E LT 0.4EV	5.88E-11	3.10	2.04E-13	3.10	2.05E-13	3.10	2.14E-13	3.10
GAMMA RAY DOSE, RAD(S CARBON)	4.72E-03	1.40	1.65E-05	1.40	1.65E-05	1.40	1.73E-05	1.40

CLEVELAND, OHIO

TITLE Radiation and Shielding Analysis Report - Nuclear Subsystem	
	DOCUMENT NO. WANL-TME-2757	DATE November, 1970
ABSTRACT This report documents the Radiation and Shielding Analysis of the NUCLEAR SUBSYSTEM (NSS) and its Engineering Critical Components (ECC). This report constitutes the primary source of all steady state and transient radiation and shielding data. The detailed Radiation and Shielding Analysis consisted of the following: Support for the derivation and establishment of appropriate performance and design requirements for the NSS and its ECs. Establishment and substantiation of the appropriate parameters of the selected design.		
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