4.2 FUEL SYSTEM DESIGN

The fuel system includes the fuel assembly (channeled fuel bundle) and the portion of the control rod assembly which extends above the coupling mechanism on the control rod drive. The following sections discuss the thermal/mechanical design bases, design descriptions, and design evaluations for the fuel system components. Nuclear design is described in Section 4.3 and thermal hydraulic design is described in Section 4.4.

4.2.1 Design Bases

4.2.1.1 Fuel Assembly

The core designs described in Section 4.3 contain two fuel assembly designs. Beginning with cycles U2C21 and U1C23, the Framatome ATRIUM-11 is the primary fuel type loaded into the core. During the fuel type transition cycles including U1C23, U1C24, U2C21, and U2C22, the cores will contain both the Framatome ATRIUM-10 and ATRIUM-11 fuel types. Occasionally, Lead Use Assemblies (LUAs) are also used in the core to provide operating experience with alternative fuel designs. When used, LUAs are loaded in non-limiting locations in the core.

Framatome ATRIUM-11 Fuel

The mechanical design for the ATRIUM-11 assembly is based on compliance with generic mechanical design criteria established by Framatome and approved by the NRC in Reference 4.2.6-10.

In accordance with the requirements of the approved mechanical design criteria, the ATRIUM-11 mechanical analyses were performed to provide the following assurances.

1) The fuel assembly shall not fail as a result of normal operation and AOO’s.

2) Fuel assembly damage shall never prevent control rod insertion when it is required.

3) The number of fuel rod failures shall be conservatively estimated for postulated accidents.

4) Fuel coolability shall always be maintained.

Mechanical design analyses have been performed to evaluate the cladding stress and strain limits, fretting wear, oxidation, hydriding and crud buildup, fuel rod bowing, differential fuel rod growth, internal hydriding, cladding collapse, and cladding and fuel pellet overheat. The RODEX4 code (Reference 4.2.6-7) was used in the mechanical design analyses.

Mechanical analyses have also been performed to evaluate the ATRIUM-11 fuel design for Seismic/LOCA loads and for normal shipping and handling. In addition, Framatome ATRIUM-11 fuel has been evaluated for power uprate conditions.

Results of these analyses and evaluations are discussed in Section 4.2.3.

Framatome ATRIUM-10 Fuel

The mechanical design for the ATRIUM-10 assembly is based on compliance with generic mechanical design criteria established by Framatome and approved by the NRC in Reference 4.2.6-10.

In accordance with the requirements of the approved mechanical design criteria, the ATRIUM-10 mechanical analyses were performed to provide the following assurances.

1) The fuel assembly shall not fail as a result of normal operation and AOO's.

2) Damage to fuel assemblies shall never prevent control rod insertion when required.

3) The number of fuel rod failures is not underestimated for postulated accidents.

4) Fuel coolability shall always be maintained.

Mechanical design analyses have been performed to evaluate the cladding stress and strain limits, fretting wear, oxidation, hydriding and crud buildup, fuel rod bowing, differential fuel rod growth, internal hydriding, cladding collapse, and cladding and fuel pellet overheat. The RODEX2, RODEX2A, RAMPEX, and COLAPX codes were used in the mechanical design analyses.

Mechanical analyses have also been performed to evaluate the ATRIUM-10 fuel design for Seismic/LOCA loads and for normal shipping and handling. In addition, Framatome ATRIUM-10 fuel has been evaluated for power uprate conditions.

Results of these analyses and evaluations are discussed in Section 4.2.3.

Lead Use Assemblies (LUA)

Occasionally, LUAs are loaded into the reactor core. The core loading of the LUAs will be such that the assemblies are not loaded into thermally limiting locations. Mechanical design analyses are performed for the LUA to evaluate fuel design parameters similar to that provided for the reload fuel.

4.2.1.2 Original Equipment Control Rod Assembly

The design bases for the original equipment control rod assembly are presented in Reference 4.2.6-3. The End-of-Life evaluation for the original equipment control rod assembly was modified in accordance with Susquehanna Nuclear, LLC's (Susquehanna Nuclear) response to IE Bulletin 79-26, Rev. 1. Susquehanna Nuclear has committed to replacing these control rod assemblies prior to exceeding a limit of 34% B^{10} depletion averaged over the upper one-fourth of the control rod assembly.

4.2.1.3 GE Duralife 160C Control Rod Assembly

The Duralife 160C control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Duralife 160C control rod assembly design bases have been reviewed and approved by the NRC (References 4.2.6-4 and 4.2.6-5).

4.2.1.4 GE Marathon Control Rod Assembly

The Marathon control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Marathon control rod assembly design bases have been reviewed and approved by the NRC (References 4.2.6-12 and 4.2.6-16).

4.2.1.5 Westinghouse CR 99 Control Rod Assembly

The Westinghouse CR 99 control rod has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Westinghouse CR 99 control rod assembly design bases have been reviewed and approved by the NRC (Reference 4.2.6-17).

4.2.2 General Design Description

A summary of fuel design characteristics is provided in Table 4.2-14 for fuel designs loaded in the core at SSES.

4.2.2.1 Core Cell

A core cell consists of a control rod assembly and the four fuel assemblies which immediately surround the control rod. Figure 4.2-15 provides nominal dimensions for a core cell loaded with ATRIUM-10 fuel and a Duralife 160C control rod. Figures 4.2-15A and 4.2.15B provide nominal dimensions for a core cell loaded with ATRIUM-10 fuel and a Marathon C+ control rod and a Marathon Ultra – HD control rod, respectively. Figure 4.2-15C provides the nominal dimensions for a core cell loaded with ATRIUM-10 fuel and a Westinghouse CR 99 control rod. Figure 4.2-16A provides nominal dimensions for a core cell loaded with ATRIUM-11 fuel and a typical GE Hitachi control rod. Figure 4.2-16B provides nominal dimensions for a core cell loaded with ATRIUM-11 fuel and a typical Westinghouse control rod. These figures illustrate the general layout of a core cell while providing nominal dimensions for fuel and control rods. A core cell may contain multiple fuel types, regardless of control rod type utilized.

Each core cell is associated with a four-lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces.

4.2.2.2 Fuel Bundle

Framatome ATRIUM-11 Fuel

A Framatome ATRIUM-11 fuel bundle contains 92 full length rods, 8 long part length fuel rods, and 12 short part length fuel rods. The 20 part length fuel rods are provided to decrease the two phase pressure loss in the top of the bundle thereby providing fuel bundle design that is more stable. A central water channel, which displaces a 3x3 array of fuel rods in the center of the bundle, provides additional moderation within the bundle thereby enhancing fuel utilization. The water channel is also used to fix the spacer locations within the fuel bundle and serves as the main structural member connecting the upper and lower tie plates. A total of 9 spacers are used to maintain fuel rod spacing. Reference 4.2.6-13 provides detailed discussion of the various components of the ATRIUM-11 fuel bundle. Nominal dimensions for the ATRIUM-11 fuel bundle are provided in Figures 4.2-16A and 4.2-16B. A schematic of an ATRIUM-11 fuel bundle is shown in Figure 4.2-17-1.

Framatome ATRIUM-10 Fuel

A Framatome ATRIUM-10 fuel bundle contains 83 full length and 8 part length fuel rods. The 8 part length fuel rods are provided to decrease the two phase pressure loss in the top of the bundle thereby providing fuel bundle design that is more stable. A central water channel, which displaces a 3x3 array of fuel rods near the center of the bundle, provides additional moderation within the bundle thereby enhancing fuel utilization. The water channel is also used to fix the spacer locations within the fuel bundle and serves as the main structural member connecting the upper and lower tie plates. A total of 8 spacers are used to maintain fuel rod spacing. Reference 4.2.6-11 provides detailed discussion of the various components of the ATRIUM-10 fuel bundle. Nominal dimensions for the ATRIUM-10 fuel bundle are provided in Figure 4.2-15. A schematic of an ATRIUM-10 fuel bundle is shown in Figure 4.2-17.

4.2.2.3 Fuel Assembly

A fuel assembly is a fuel bundle including the surrounding fuel channel. The fuel assemblies are arranged in the reactor core to approximate a right circular cylinder inside the core shroud. Each fuel assembly is supported by a fuel support piece and the top guide.

The fuel channel enclosing the Framatome fuel bundles is fabricated from one of three alloys: Framatome's proprietary Z4B, Zircaloy-4, or Zircaloy-2. Fuel channels perform the following functions: the channel separates flow inside the bundle from the bypass flow between channels; the channel guides the control rod and provides a bearing surface for it; the channel provides rigidity for the fuel bundle. Fuel channels are attached to ATRIUM-11 fuel bundles by two bolts on the upper tie plate, and to ATRIUM-10 fuel bundles by channel fasteners. The ATRIUM-11 channel fastener is integral to the ATRIUM-11 fuel channel while the ATRIUM-10 channel fastener is a separate component that attaches the ATRIUM-10 fuel channel to the ATRIUM-10 fuel bundle using a threaded post on the upper tie plate. Both the ATRIUM-10 and ATRIUM-11 channel fasteners have springs which helps to hold the assembly against the core grid. Once the channel fastener has been installed and the fuel assembly has been positioned in the core, the spring on the channel fastener helps to hold the assembly against the core grid. Schematics of typical ATRIUM-10 and ATRIUM-11 fuel channels are shown in Figures 4.2-18, 4.2-18-1, 4.2-18-2, 4.2-18-3, and 4.2-18-4.

The Advanced Fuel Channel (AFC) has been introduced on fuel for Units 1 and 2 beginning with the reload for U1C20 and is depicted in Figures 4.2-18-3 and 4.2-18-4. The AFC has thinned side-walls in the active core region. Details regarding the ATRIUM-10 AFC are available in Reference 4.2.6-8. Details regarding the ATRIUM-11 AFC are available in Reference 4.2.6-13.

The ATRIUM-11 fuel assemblies use the Framatome 3rd Generation FUELGUARD (3GFG) lower tie plate (LTP) design. The 3GFG LTP uses a modular approach to integrate the inlet nozzle, the structural grid and the filtering elements of the lower tie plate assembly, making the filter a non-load bearing component. The inlet transition nozzle, the structural grid and the filtering elements of the LTP are made out of 304L stainless steel. The modular design supports separation of filter function from structural function. The 3GFG filter insert design provides a higher level of filtering efficiency protection against the entry of shorter and smaller debris than the previous ATRIUM-10 standard FUELGUARD.

The first reload of ATRIUM-11 fuel assemblies for U2C21 utilizes the “Optimized Load Chain” (OLC) design. This design is similar to that which has been used the ATRIUM-10 fuel type.

Proper assembly orientation in the core is verified by visual inspection and is assured by verification procedures during core loading. Five visual indications of proper fuel assembly orientation exist. These indications are:

1) The channel fastener assemblies are located at one corner of each fuel assembly adjacent to the center of the control rod. The ATRIUM-11 channel fastener has an orientation hole, and the ATRIUM-10 channel fastener has a bolt.

2) The orientation boss on the fuel assembly handle points toward the adjacent control rod.

3) The channel spacing buttons are adjacent to the control rod passage area.

4) The assembly identification numbers which are located on the fuel assembly handles are all readable from the direction of the center of the cell.

5) There is cell-to-cell replication.

Proper assembly orientation in the core is shown in Figure 4.2-19.

4.2.2.4 Reactivity Control Assembly

4.2.2.4.1 Original Equipment Control Rod Assembly

The control rod consists of a sheathed cruciform array of commercial grade stainless steel tubes filled with B_4C powder. The main structural member of a control rod is made of Type-304 stainless steel and consists of a top handle, a bottom casting with a velocity limiter and control rod drive coupling, a vertical cruciform center post, and four U-shaped absorber tube sheaths. Control rods are cooled by core bypass flow.

Stellite rollers with Haynes Alloy 25 pins located at the top and bottom of the control rod help guide the control rod as it is inserted and withdrawn from the core.

The control rod velocity limiter consists of cast austenitic stainless steel (Grade CF-8) and is an integral part of the control rod bottom casting. The velocity limiter protects against high reactivity insertion rate by limiting the control rod velocity in the event of a control rod drop accident.

Reference 4.2.6-3 provides the general design characteristics of the original equipment control rod assembly. A diagram of this control rod is provided in Figure 4.2-20.

4.2.2.4.2 Duralife 160C Control Rod Assembly

The main differences between the Duralife 160C control rods and the original equipment control rods are:

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

- the Duralife 160C control rods utilize improved B_4C tube material (i.e. high purity type 304 stainless steel vs. commercial purity stainless steel) to eliminate cracking in the remaining B_4C rods during the lifetime of the control rod;

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

- the Duralife 160C control rods utilize low cobalt-bearing pin and roller materials in place of stellite which was previously used (PH13-8 Mo for pins, Inconel X750 for rollers);

- the Duralife 160C control rod handles are longer by approximately 3.1 inches. The extended handle provides lateral support against the top grid and facilitates fuel moves within the reactor vessel during refueling outages;

- the Duralife 160C control rods are roughly 10% to 15% heavier (depending on the velocity limiter design) as a result of the design changes described above; and

- the Duralife 160C control rod velocity limiter material is cast austenitic stainless steel grade CF-3.

References 4.2.6-4 and 4.2.6-5 provide additional discussion on the design of the various Duralife control rod assemblies, including the features of the Duralife 160C inserted in the SSES Units. A cross section of the Duralife 160C control rod is shown in Figure 4.2-15, and a diagram of this control rod is provided in Figure 4.2-21.

4.2.2.4.3 GE Marathon Control Rod Assembly

The main difference between the Marathon control rod and the Duralife design control rods are:

- the absorber tube and sheath arrangement of the Duralife designs is replaced with an array of square tubes resulting in reduced weight and increased absorber volume;

- the full length center tie rod is replaced with a segmented tie rod which also reduces weight.

References 4.2.6-12 and 4.2.6-16 provide additional discussion on the designs of the Marathon control rod assemblies. A cross section of the Marathon C+ control rod is shown in Figure 4.2-15A. A cross section of the Marathon Ultra HD Control Rod is shown in Figure 4.2-15B. A diagram of the control rod is provided in Figure 4.2-22.

Replacement Marathon control rods may use a modified handle assembly that eliminates pins and rollers present in the earlier design. Figure 4.2-22 is applicable to the rollerless design; the only difference is that the small circles depicting the rollers would be removed from the diagram for the rollerless design. The overall shape and dimensions of the upper handle remains unchanged.

4.2.2.4.4 Westinghouse CR 99 Control Rod Assembly

The Westinghouse BWR CR 99 control rod design is comparable to that of the GE design. Both the Westinghouse and GE control blade designs have a coupling socket, velocity limiter, coupling release handle, 143-inch active absorber zone, B_4C as their main absorber material and a handle. The overall length of the Westinghouse CR 99 control rod is the same as the OE GE control rod design at 173-inches. The main difference in the design between the Westinghouse and GE control rod is that the Westinghouse CR 99 control blade has horizontal absorber holes drilled in solid stainless steel wings. Reference 4.2.6-17 provides additional discussion on the design of the CR 99 control rod. A diagram of the CR 99 control rod is provided in Figure 4.2-23.

4.2.3 Design Evaluations

4.2.3.1 Fuel Design Evaluations

Framatome ATRIUM-11 Fuel

For the ATRIUM-11 fuel, the design is such that adequate margins to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) are assured for all anticipated operational occurrences (AOOs) throughout the life of the fuel as demonstrated by the fuel mechanical design analyses (References 4.2.6-10 and 4.2.6-15), provided that the steady state fuel rod power remains within the power history assumed in the analyses. The design steady state power history for the ATRIUM-11 fuel is shown in Reference 4.2.6-15 and is incorporated into the Unit/Cycle specific Core Operating Limits Report (COLR) as an operating limit. The operating limit may be in terms of planar or pellet exposure. ARTS has been implemented for Unit 1 and Unit 2. The COLR has a flow dependent LHGR multiplier and a power dependent multiplier, which are used to adjust the LHGR limit at off-rated conditions to assure that design limits are not exceeded. The mechanical analyses support a maximum assembly average exposure of 57,000 MWd/MTU.

Framatome has evaluated the performance of ATRIUM-11 fuel assemblies under Susquehanna Seismic LOCA conditions. For this evaluation, maximum loads and/or stresses were calculated for the fuel components under an acceleration load equivalent to a maximum dynamic load which bounds the allowable bending moment in BWR/4 Advanced Fuel Channels, (Reference 4.2.6-8). The large margin that resulted from these analyses shows that the ATRIUM-11 fuel assembly with an Advanced Fuel Channel demonstrates adequate structural integrity in the Susquehanna Units under Seismic LOCA conditions. With regard to assembly liftoff, the net force for the ATRIUM-11 fuel assembly was found to be downwards.

Framatome ATRIUM-10 Fuel

For the ATRIUM-10 fuel, the design is such that adequate margins to fuel mechanical design limits (e.g., centerline melting temperature, transient strain, etc.) are assured for all anticipated operational occurrences (AOOs) throughout the life of the fuel as demonstrated by the fuel mechanical design analyses (References 4.2.6-10 and 4.2.6-11), provided that the steady state fuel rod power remains within the power history assumed in the analyses. The design steady state power history for the ATRIUM-10 fuel is shown in Reference 4.2.6-11 and is incorporated into the Unit/Cycle specific Core Operating Limits Report (COLR) as an operating limit. The operating limit may be in terms of planar or pellet exposure. ARTS has been implemented for Unit 1 and Unit 2. The COLR has a flow dependent LHGR multiplier and a power dependent multiplier, which are used to adjust the LHGR limit at off-rated conditions to assure that design limits are not exceeded. The mechanical analyses for fresh ATRIUM-10 assemblies support a maximum assembly average exposure of 54,000 MWd/MTU.

Framatome has evaluated the performance of ATRIUM-10 fuel assemblies under Susquehanna Seismic LOCA conditions. For this evaluation, maximum loads and/or stresses were calculated for the fuel components under an acceleration load equivalent to a maximum dynamic load which bounds the allowable bending moment in BWR/4 80 mil, 100 mil, and Advanced fuel channels, (Reference 4.2.6-8). The large margin that resulted from these analyses shows that the ATRIUM-10 fuel assembly with an 80 mil, 100 mil, or Advanced fuel channel demonstrates adequate structural integrity in the Susquehanna Units under Seismic LOCA conditions. With regard to assembly liftoff, the net force for the ATRIUM-10 fuel assembly was found to be downwards.

4.2.3.2 Results of Control Rod Assembly Design Evaluations

4.2.3.2.1 Original Equipment Control Rod Assembly

The original equipment control rod assembly design evaluations are discussed in Reference 4.2.6-3. Subsequent to the completion of the above referenced evaluations, a new failure mechanism was identified for the original equipment control rod assembly. IE Bulletin 79-26 Rev-1 discusses this failure mechanism and recommends a reduction in the end-of-life criteria for the original equipment control rod assembly. Susquehanna Nuclear committed to replacing the original equipment control rod assemblies in accordance with this IE bulletin.

4.2.3.2.2 Duralife 160C Control Rod Assembly

The Duralife 160C control rod stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control rod insertion capability has been evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. The Duralife 160C control rod coupling mechanism is equivalent to the original equipment coupling mechanism, and is therefore fully compatible with the existing control rod drives in the plant. In addition, the materials used in the Duralife 160C are compatible with the reactor environment. The Duralife 160C control rods are roughly 10% to 15% heavier than the original equipment control rod assembly, depending on the velocity limiter design utilized. The impact of the increased weight of the control rods on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been evaluated and found to be negligible.

With the exception of the crevice-free structure and the extended handle, the Duralife 160C control rod is equivalent to the NRC approved Hybrid I Control Rod Assembly (Reference 4.2.6-4). The mechanical aspects of the crevice-free structure were approved by the NRC for all control rod designs in Reference 4.2.6-5. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same NRC approved nuclear interchangeability evaluation methodology as described in Reference 4.2.6-4. These calculations were performed for the original equipment control rods and the Duralife 160C control rods assuming an infinite array of Framatome 9x9-2 fuel. The Duralife 160C control rod has a slightly higher worth than the original equipment design, but the increase in worth is within the criterion for nuclear interchangeability. The increase in rod worth has been taken into account in the appropriate reload analyses.

In Reference 4.2.6-4, the NRC approved the Hybrid I (Duralife 160C) control rod which weighs less than the D lattice control rod. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control rod (including the extended handle, crevice free structure, and heavier velocity limiter) weighs less than the D lattice control rod, and the heavier D lattice control rod drop speed is used in the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control rod scram times are not significantly different than the original equipment control rod scram times.

Also, the scram speeds are monitored in the plant to assure compliance with safety analysis assumptions and technical specification limits.

IE Bulletin 79-26, Rev. 1 was issued to address B_4C rod cracking and subsequent loss of boron in GE original equipment control rods. The Duralife 160C control rod design contains solid hafnium absorber rods in locations where B_4C tubes have historically failed. The remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steel vs. commercial purity stainless steel), thus, boron loss due to cracking is not expected to occur.

Due to the control rod design, IE Bulletin 79-26, Rev. 1 does not apply to Duralife 160C control rods. However, Susquehanna Nuclear plans to continue tracking the depletion of each control rod and discharge any control rod prior to a ten percent loss in reactivity worth.

4.2.3.2.3 GE Marathon Control Rod Assembly

The form, fit and function of the Marathon control rod design are equivalent to the original equipment control rods used at Susquehanna. Reference 4.2.6-12 documents NRC acceptance of the GE Marathon control rod mechanical design.

The control rod stresses, strains, and cumulative fatigue were evaluated by GE Nuclear and result in acceptable margins to safety. The control rod insertion capability was evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. In addition, the coupling mechanism is fully compatible with the existing control rod drives in the plant. The materials used in the Marathon control rods were also evaluated and are compatible with the reactor environment. The Marathon control rods are approximately the same weight as the original equipment control rods and, therefore, there is no impact on the seismic and hydrodynamic load evaluation for the reactor vessel and internals. With lighter weight than the D160 control rods and envelope dimensions less than or the same as the original equipment, the Marathon design is compatible with existing NSSS hardware and there is no change in scram performance or drop time.

Neutronics evaluations of the Marathon control rods by GE Nuclear using the methodologies described in Reference 4.2.6-12 indicate the C lattice Marathon design for Susquehanna slightly exceeds the +5% beginning-of-life reactivity worth constraint relative to the original equipment all B_4C design. Therefore, the effect of the increased reactivity worth on plant analyses had to be considered. The increased reactivity worth was found to not adversely impact normal operation, and is considered in the analysis of abnormal operational occurrences, infrequent events, or accidents. The Marathon Ultra – HD control rod design satisfies the +5% beginning-of-life reactivity worth constraint, (Reference 4.2.6-16).

The Marathon control rods used improved materials and contain significant design improvements to eliminate cracking and the associated loss of Boron experienced by the original equipment. GE defines the end of life as a 10% reduction in cold reactivity worth in any ¼ axial segment relative to the initial undepleted state of the original equipment control rods. Susquehanna Nuclear will track the depletion of the Marathon control rods and discharge any control rod prior to reaching the defined end of life, or provide technical justification for its continued use.

4.2.3.2.3 Westinghouse CR 99 Control Rod Assembly

The form, fit and function of the Westinghouse CR 99 control rod design are equivalent to the OE control rods used at Susquehanna. Reference 4.2.6-17 documents NRC acceptance of the CR 99 control rod mechanical design.

The CR 99 control rod stresses, strains, and cumulative fatigue were evaluated by Westinghouse and result in acceptable margins to safety. The CR 99 insertion capability was evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. The CR 99 coupling mechanism is fully compatible with the existing control rod drives in the plant. The materials used in the CR 99 control rods were also evaluated and are compatible with the reactor internals and the reactor environment. The CR 99 control rods are similar in nominal weight of the OE control rods and, therefore, there is no impact on the seismic and hydrodynamic load evaluation for the reactor vessel and internals. Scram speeds and settling times in the reactor are not adversely affected by the CR 99 control rods. The CR 99 velocity limiter design is identical to the design of the OE control rods and meets the assumption for the control rod drop accident.

The total worth of the CR 99 control rod is within ±5% of the OE control rod. There is no negative impact to shutdown margin and minimal impact on LPRM detector indications. The nuclear end of life criteria is maintained as 10% reactivity worth decrease relative to the OE control rod (Reference 4.2.6-17).

The CR 99 use of an improved high density absorber material, which is less sensitive to both powder densification and absorber swelling due to neutron absorption reactions, minimizes the possibility of absorber swelling causing contact with the surrounding stainless steel and contributing stress. The CR 99 use of AISI 316L stainless steel, with its better resistance to fast neutron irradiation assisted stress corrosion cracking (IASCC), also reduces the potential for control blade cracking. SSES will track the depletion of the CR 99 control rod and discharge any control rods prior to reaching the defined end of life, or provide technical justification for its continued use.

4.2.4 Testing and Inspection

4.2.4.1 Fuel Hardware and Assembly

Framatome has developed Quality Control Standards for manufacturing, testing, and inspection of components and fuel bundles.

On-site inspection of all new fuel bundles, fuel channels, and control rods is performed prior to installation into a Susquehanna Unit. These inspections are controlled by plant procedures. The procedures were developed based on guidelines provided by the fuel, channel, and control rod suppliers for receipt inspection and include acceptance criteria which are verified for each fuel bundle, fuel channel, or control rod.

The fuel channel management practices in place at Susquehanna are consistent with the recommendations contained in GE SIL 320, 'Recommendations for Mitigation of the Effects of Fuel Channel Bowing'. In addition, Susquehanna Nuclear will only use fuel channels for only one fuel bundle lifetime and will not reuse them. The fuel channel management practices are continuously reviewed against plant operation and industry practices.

4.2.4.2 Enrichment, Burnable Poison, and Absorber Rod Concentrations

Framatome has established adequate measures, in accordance with their Quality Assurance Program, to assure that nuclear materials of varying enrichment and form are positively identified and physically segregated as required to assure no inadvertent intermixing of enrichment forms. These measures include, as appropriate, identification of storage and processing containers, gamma scan verification of powder, nuclear rod assay, analytical examinations, in-process inspections, cleanouts of processing equipment between enrichments, administrative controls on the handling of materials, and audits of processing and product.

Framatome fuel pellets are manufactured in accordance with approved procedures and are controlled by Product Design Specifications which define the allowable concentration tolerances and confidence levels required to verify enrichment and burnable poison concentrations.

General Electric (GE) supplied the original equipment control rods for both Susquehanna Units and is the supplier of the Duralife-160C and Marathon replacement control rods. The absorber materials, boron carbide and hafnium, are certified by GE to meet GE Material Specifications. The isotopic B^{10} content and boron content is verified for each powder lot received by General Electric. All boron carbide absorber rod assemblies are subjected to a leakage test to insure absorber rod integrity. GE performs analysis of hafnium absorber rod lots to insure chemical composition is in conformance with GE Material Specifications.

Westinghouse supplied the CR 99 control rods for use at both Susquehanna Units. The boron carbide absorber material is certified by Westinghouse to meet Westinghouse Material

Specifications. The isotopic B^{10} content and boron content is verified for each powder lot received by Westinghouse. Each CR 99 control rod blade is leak tested with helium per the Westinghouse Materials Specifications.

4.2.4.3 Surveillance, Inspection, and Testing

Susquehanna Nuclear has a fuel reliability program that includes fuel performance monitoring and fuel failure response. On-line fuel performance monitoring is conducted to determine whether there is a fuel failure and may include evaluation of the general location of the failed assembly, the number of fuel assemblies suspected, when the failure occurred, and the approximate exposure of the failed assembly. Determination of this information prior to refueling allows preparation for changes in the following cycle's core design. In addition, control rod sequence exchanges and full power control rod patterns can be developed to minimize the offgas release from the failed rod(s) and stress on the suspect assembly during power maneuvering.

On-line fuel performance monitoring is performed at the Susquehanna station by periodic evaluation of pretreatment offgas activity and/or reactor coolant samples. Verification of failed fuel is made by periodic evaluation of the pretreatment xenon and krypton offgas activity and reactor water cleanup system iodine and cesium activity. The general location of the failed fuel assembly is, typically, identified by control rod motion testing and monitoring of the pretreatment offgas activity. Identification of the exact assembly may be performed by sipping or ultrasonic testing (UT) of the suspect assemblies.

Post-irradiation fuel failure evaluations are, typically, performed to determine the exact fuel rod location within the assembly and the root cause of a fuel rod failure. The exact location of the failed rod may be determined by UT or eddy current testing. Root cause evaluations may include review of manufacturing and inspection records, visual examination of the failure location, and destructive examination of the failed fuel rod.

Post-irradiation inspection programs have been developed by Framatome to evaluate fuel design performance. Reference 4.2.6-11 discusses the Framatome inspection and surveillance program for irradiated ATRIUM-10 fuel. Reference 4.2.6-15 discusses the Framatome inspection and surveillance program for irradiated ATRIUM 11 fuel.

4.2.5 Operating and Developmental Experience

Framatome ATRIUM-11 will begin being utilized at SSES beginning with Unit 1 Cycle 23 and Unit 2 Cycle 21. Framatome ATRIUM-10 fuel has been utilized at SSES beginning with Unit 1 Cycle 11 and Unit 2 Cycle 9.

Susquehanna Nuclear continually tracks the performance of all fuel in the Susquehanna Units in an effort to identify indications of potential fuel rod failures.

Prior to the implementation of a mechanical fuel design into either Susquehanna Unit, that introduces features not currently in other operating plants, a plan will be developed to evaluate the performance of this fuel design in the Susquehanna Units. This plan may include pre and post irradiation fuel assembly characterization, visual inspection, power maneuvering evaluations, fuel clad corrosion evaluations, and UT inspections.

Susquehanna Nuclear occasionally participates in Lead Use Assembly programs. These programs allow the company to evaluate and gain operating experience with new fuel designs.

4.2.6 References

4.2.6-1 Deleted

4.2.6-2 Deleted

4.2.6-3 "BWR/4 and BWR/5 Fuel Design," NEDE-20944(P), General Electric Company, October 1976, and Letter from Olan D. Parr (NRC) to Dr. G. G. Sherwood (GE), "Review of General Electric Topical Report NEDE-20944-P, BWR/4 and BWR/5 Fuel Design (NEDO-20944 Non-Proprietary Version)", September 30, 1977.

4.2.6-4 "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly for the BWR 4/5 C Lattice, "NEDE-22290-A, General Electric Company, September 1983, and Supplement 1, General Electric Company, July 1985.

4.2.6-5 "Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly, " NEDE-22290-A, Supplement 3, General Electric Company, May 1988.

4.2.6-6 Deleted

4.2.6-7 "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors." BAW-10247PA and Supplement 2P-A.

4.2.6-8 "Mechanical Design for BWR Fuel Channels", EMF-93-177 (P) (A) Rev. 1 and Supplements 1P-A Rev. 0 and 2P Rev. 1.

4.2.6-9 Deleted

4.2.6-10 "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98(P)(A), Rev. 1 and Rev. 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

4.2.6-11 "Mechanical Design Evaluation for Siemens Power Corporation ATRIUM™ –10- BWR Reload Fuel, " EMF-95-52(P), Rev. 2, Siemens Power Corporation – Nuclear Division, December 1998.

4.2.6-12 "GE Marathon Control Rod Assembly, " NEDE-31758P-A, GE Nuclear Energy, October 1991.

4.2.6-13 "Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies," ANP-3762P.

4.2.6-14 Deleted

4.2.6-15 "Fuel Design Evaluation for ATRIUM 11 BWR Reload Fuel Information Design Report," ANP-3653P.

4.2.6-16 NEDE - 33284 Supplement 1 P-A, Revision 1 March 2012, Licensing Topical Report, “Marathon Ultra Control Rod Assembly”.

4.2.6-17 WCAP-16182-P-A, Revision 1 October 2009, “Westinghouse BWR Control Rod CR 99 Licensing Report – Update to Mechanical Design Limits”.

TABLE 4.2-14

FUEL DESIGN CHARACTERISTICS (Nominal)

PARAMETER	ATRIUM-10	ATRIUM-11
Fuel Assembly		
Fuel rod array	10x10	11x11
Fuel Rods per Assembly	91	112
Overall length (in)	176.4	176.4
Number of Spacers	8	9
Fuel Channel Wall Thickness (mils)	80, 100, or 100/75	100/71/51
Channel inside width (in)	5.278	5.283
Full Length Fuel Rod		
Number per assembly	83	92
Clad O.D. (in)	0.396	0.370
Cladding Material	Zircaloy-2	Zircaloy-2
Part Length Fuel Rod		
Number per assembly	8	N/A
Cladding Material	Zircaloy-2	N/A
Short Part Length Fuel Rod		
Number per assembly	N/A	12
Cladding Material	N/A	Zircaloy-2
Long Part Length Fuel Rod		
Number per assembly	N/A	8
Cladding Material	N/A	Zircaloy-2
Fuel Pellet		
Material	UO_2	UO_2-Cr_2O_3
Density (% Theoretical)	95.4	97.35
Chromia Content (µg/g U)	N/A	1250
Diameter (in)	0.3413	0.3193
Length (in)	0.413	0.386
Burnable Absorber	Gd_2O_3	Gd_2O_3
Water Rod/Channel		
# of Water Rods/Channels	1 channel	1 channel

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Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM™-10 FUEL WITH DURALIFE 160C CONTROL ROD) FIGURE 4.2-15

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM^{TM}-10 FUEL WITH MARATHON CONTROL ROD)
FIGURE 4.2-15A

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM^{TM}-10 FUEL WITH ULTRA-HD CONTROL ROD) FIGURE 4.2-15B

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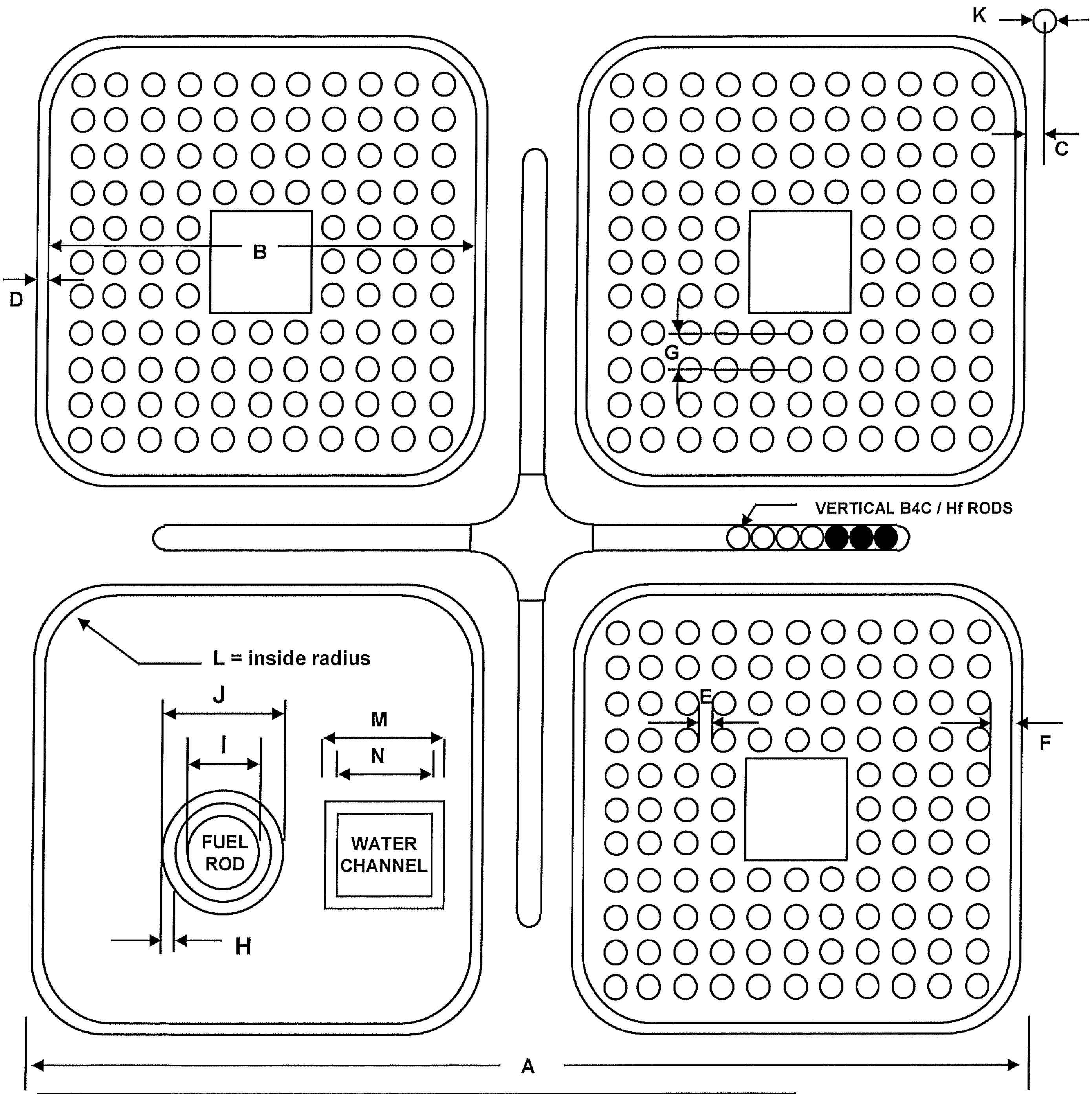
Figure Withheld Under 10 CFR 2.390

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CORE CELL (ATRIUM^{TM}-10 FUEL WITH WESTINGHOUSE CR 99 CONTROL ROD) FIGURE 4.2-15C



DIM. IDENTIFICATION	A	B	C	D	E	F	G	H	I
DIM. INCHES	12.0	NOTE	0.259	NOTE	0.100	0.209	0.470	0.022	0.319

DIM. IDENTIFICATION	J	K	L	M	N
DIM. INCHES	0.370	0.750	0.409	1.299	1.220

NOTE:
For Advanced Channel:
Dimension B = 5.283 inches in un-milled sections, and 5.323 inches at the inner milling zone
Dimension E = 0.100 inches at corners, 0.071 inches at lower 60% of channel, and 0.051 inches at upper 40% of channel

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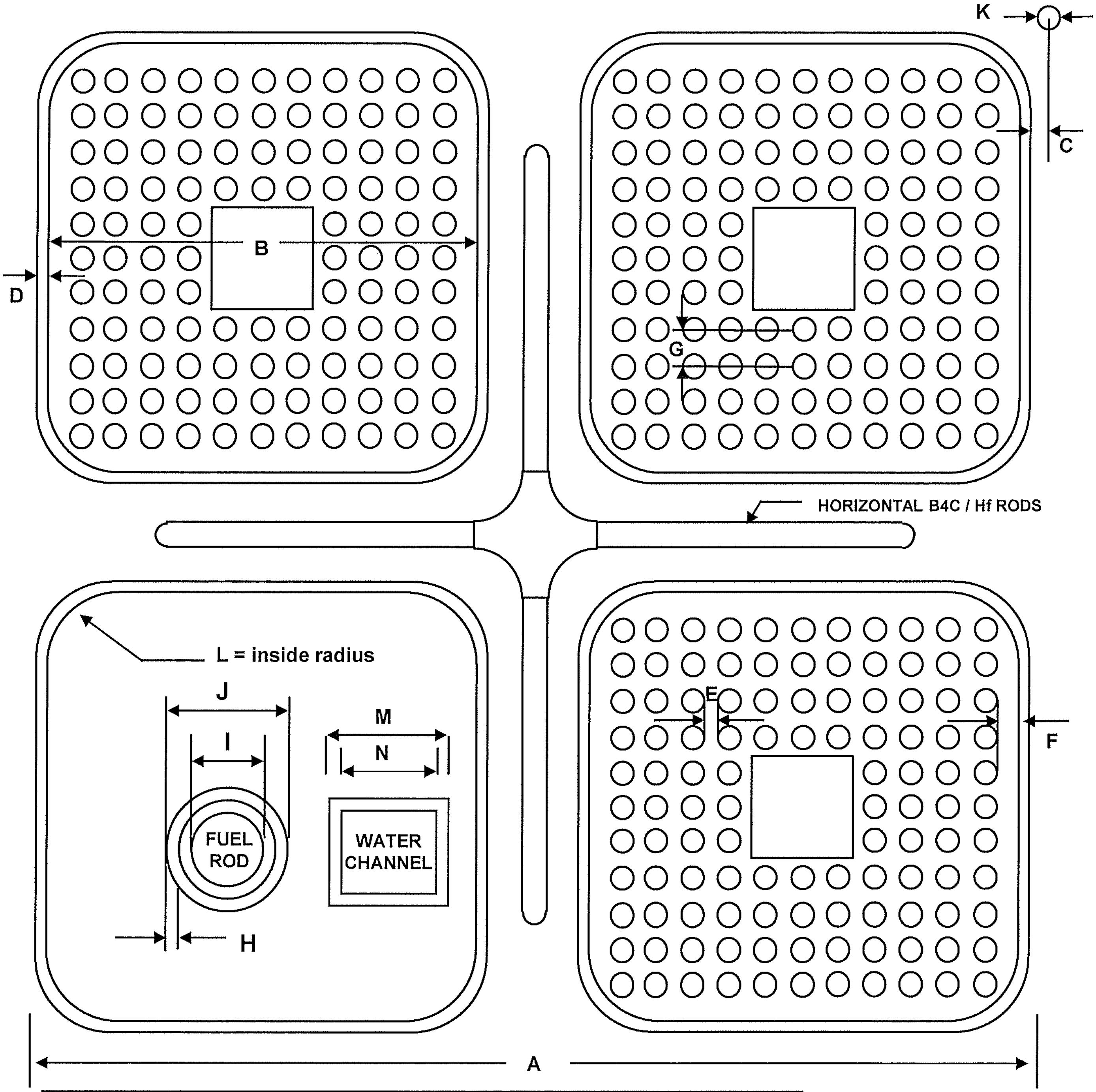
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL CORE CELL
(ATRIUM 11 FUEL WITH
GE HITACHI CONTROL ROD)

FSAR FIGURE 4.2-16A, Rev. 0

NUCLEAR FUELS



DIM. IDENTIFICATION	A	B	C	D	E	F	G	H	I
DIM. INCHES	12.0	NOTE	0.259	NOTE	0.100	0.209	0.470	0.022	0.319

DIM. IDENTIFICATION	J	K	L	M	N
DIM. INCHES	0.370	0.750	0.409	1.299	1.220

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NOTE:

For Advanced Channel:
Dimension B = 5.283 inches in un-milled sections, and 5.323 inches at the inner milling zone
Dimension E = 0.100 inches at corners, 0.071 inches at lower 60% of channel, and 0.051 inches at upper 40% of channel

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

TYPICAL CORE CELL
(ATRIUM 11 FUEL WITH
WESTINGHOUSE CONTROL ROD)

FSAR FIGURE 4.2-16B, Rev. 0

NUCLEAR FUELS

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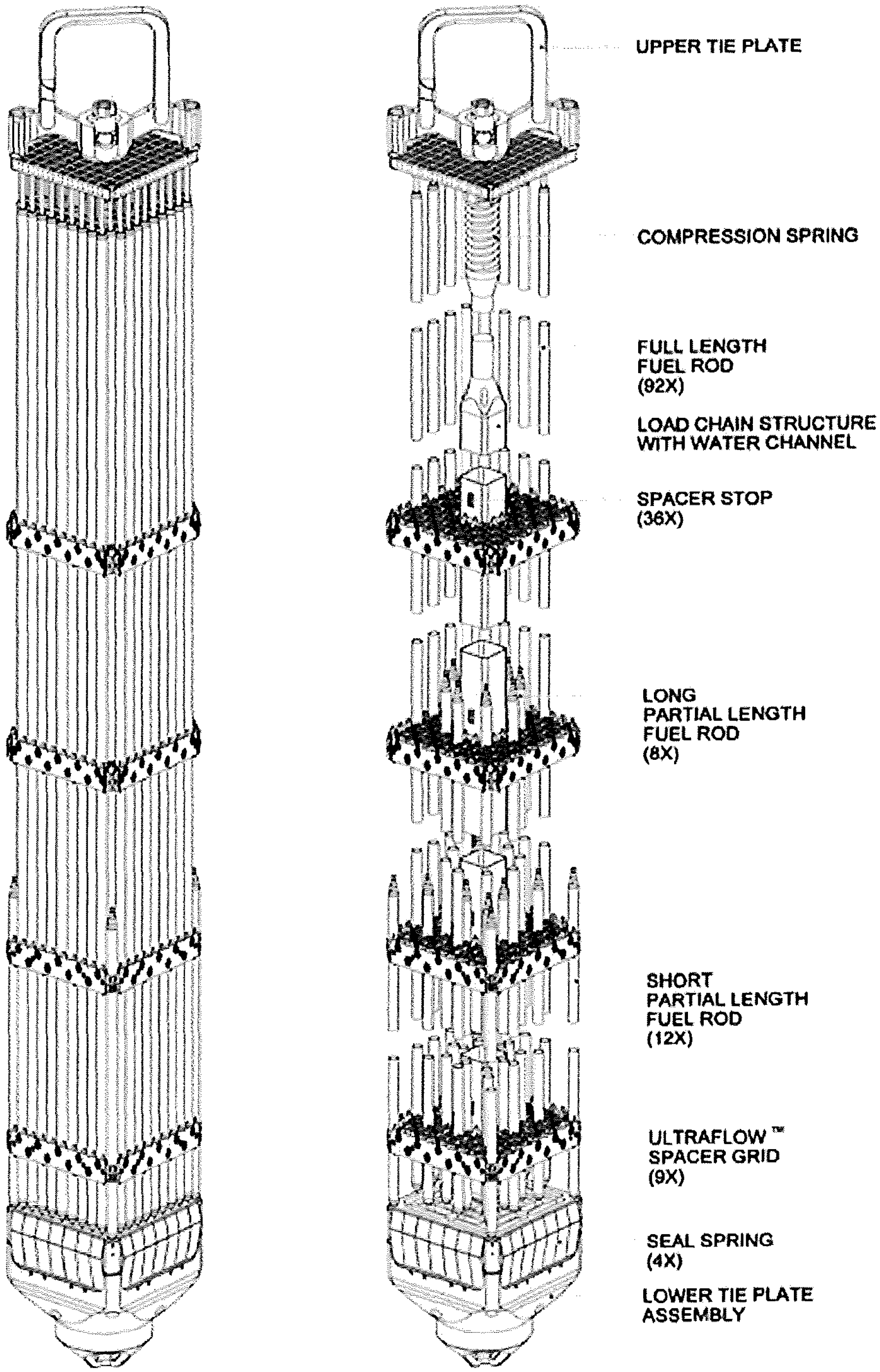
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

FUEL BUNDLE
FANP ATRIUM^{TM}-10

FIGURE 4.2-17



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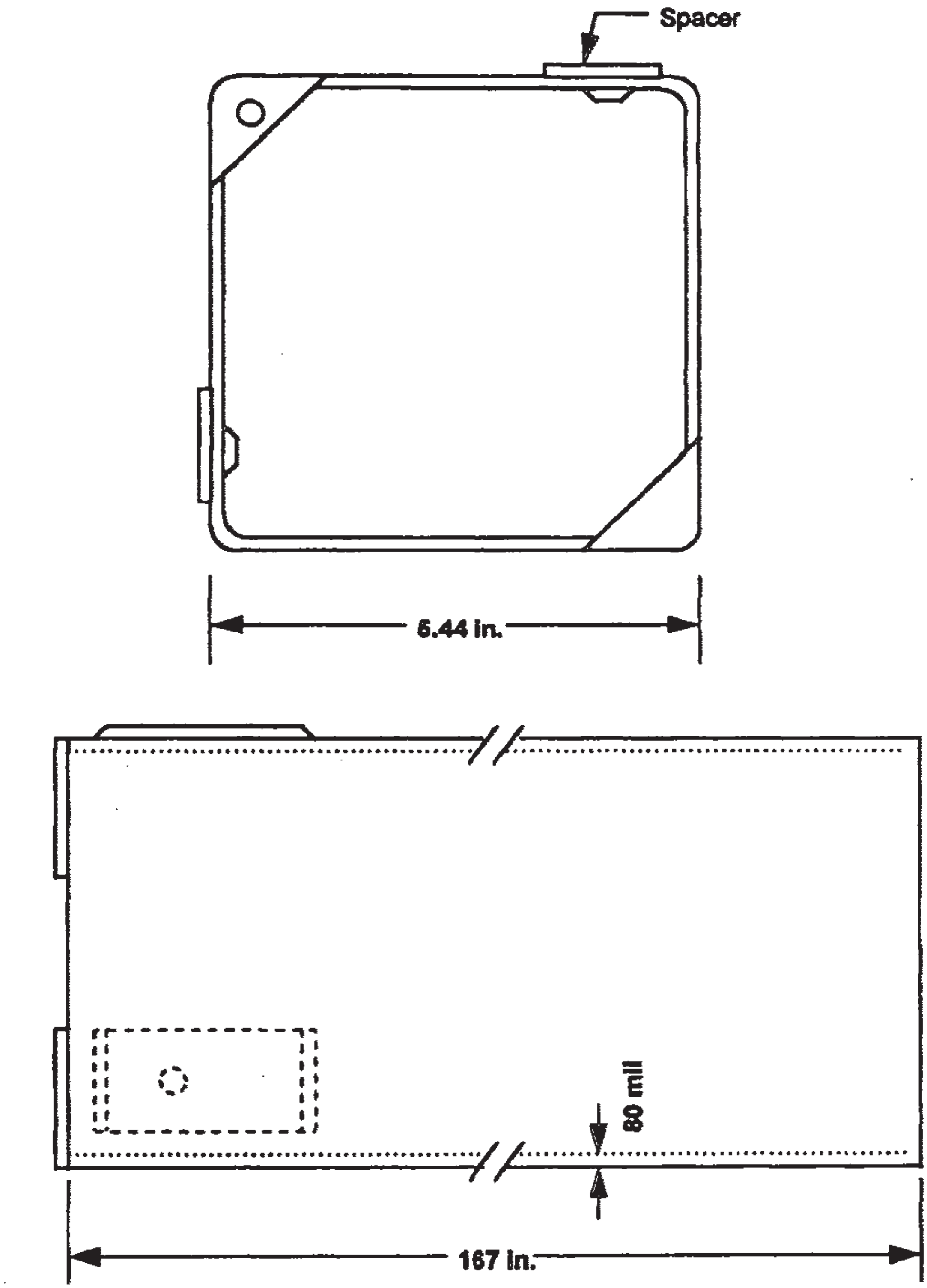
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
REPRESENTATIVE ATRIUM 11 FUEL BUNDLE (NOT TO SCALE)
FSAR FIGURE 4.2-17-1, Rev. 0



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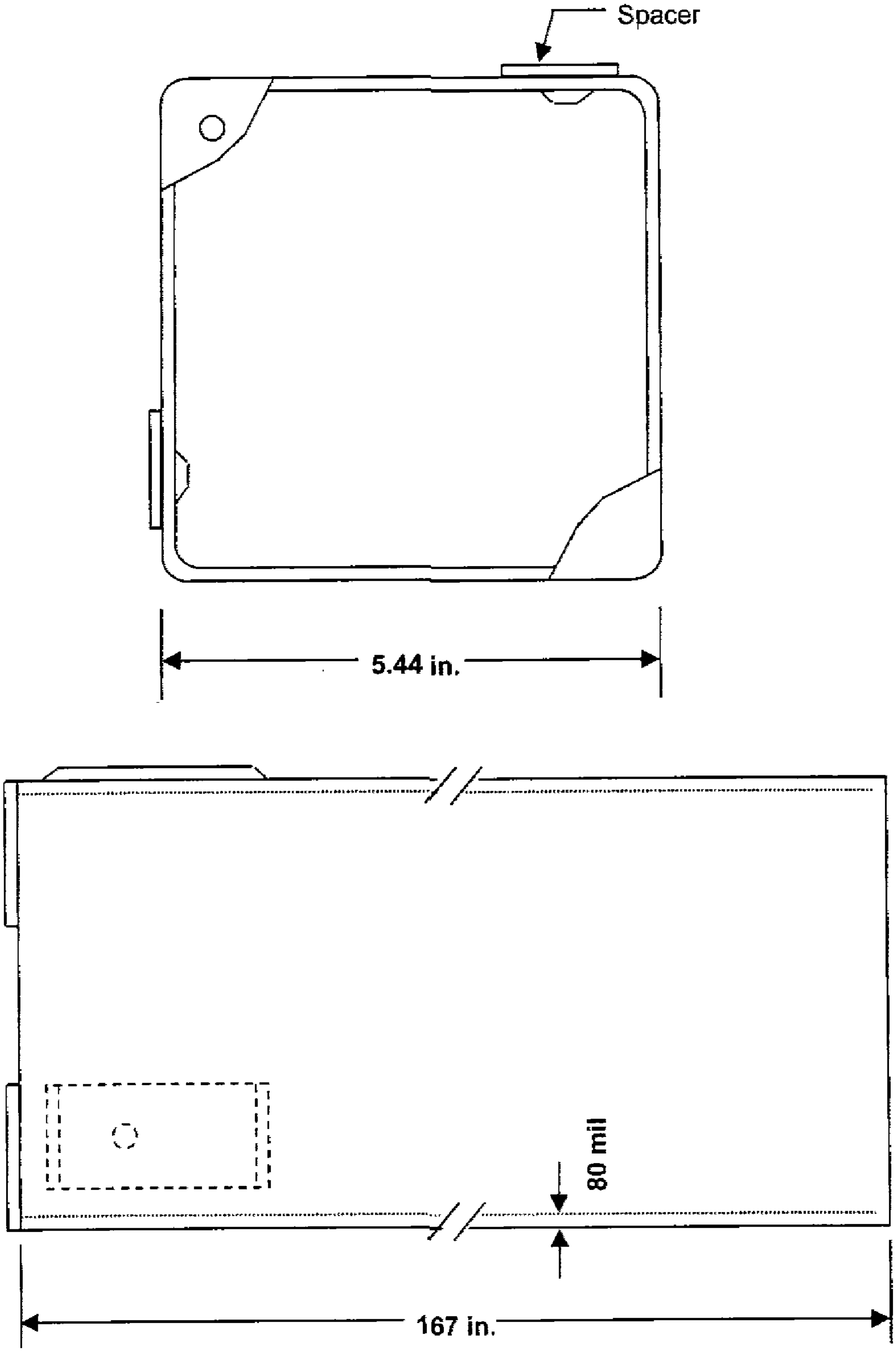
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

80 MIL FUEL CHANNEL

FIGURE 4.2-18, Rev. 54

Auto Cad: Figure Fsar 4_2_18.dwg



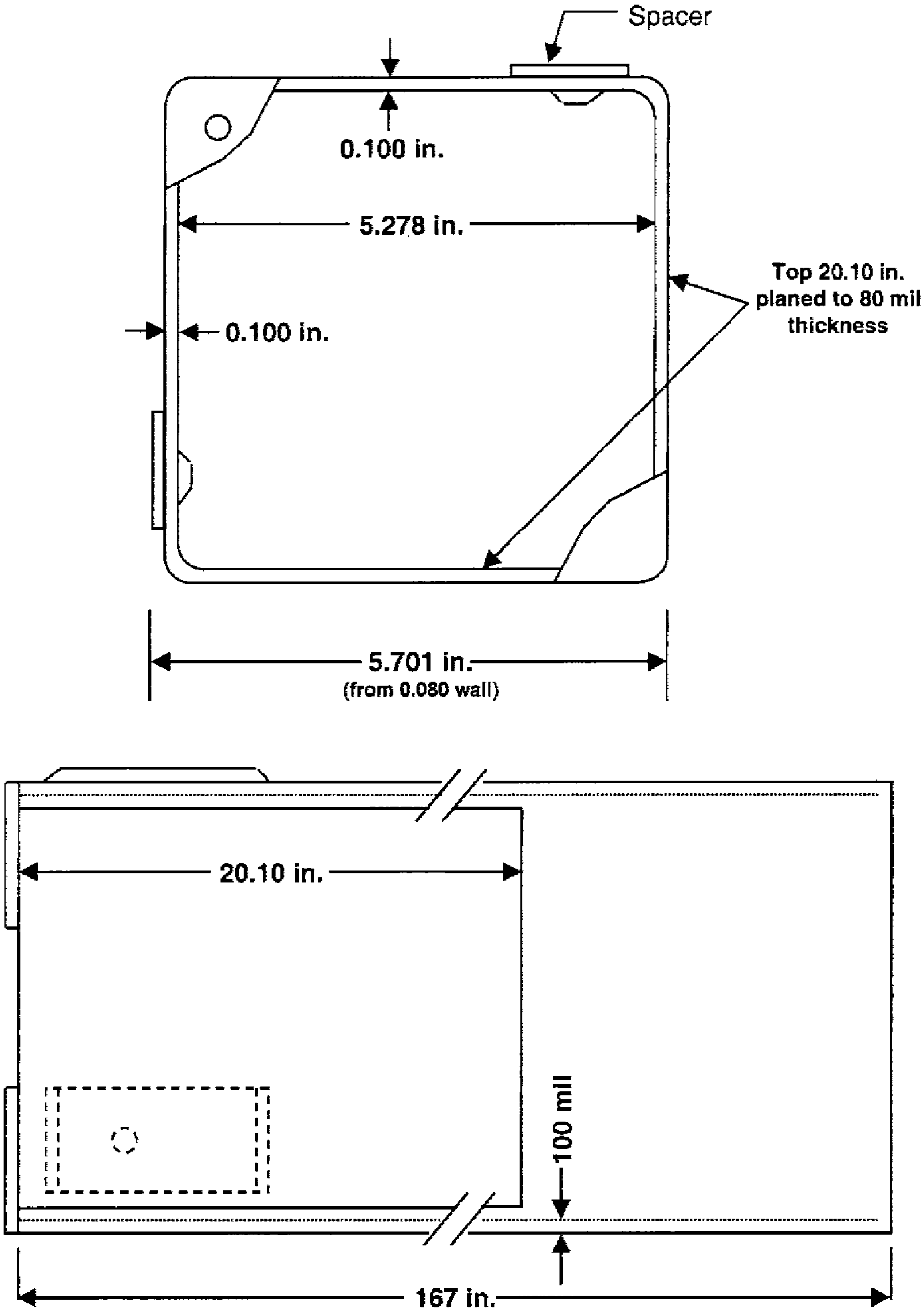
FSAR REV. 65

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

80 MIL FUEL CHANNEL

FIGURE 4.2-18-1, Rev. 1

Auto Cad: Figure Fsar 4_2_18_1.dwg



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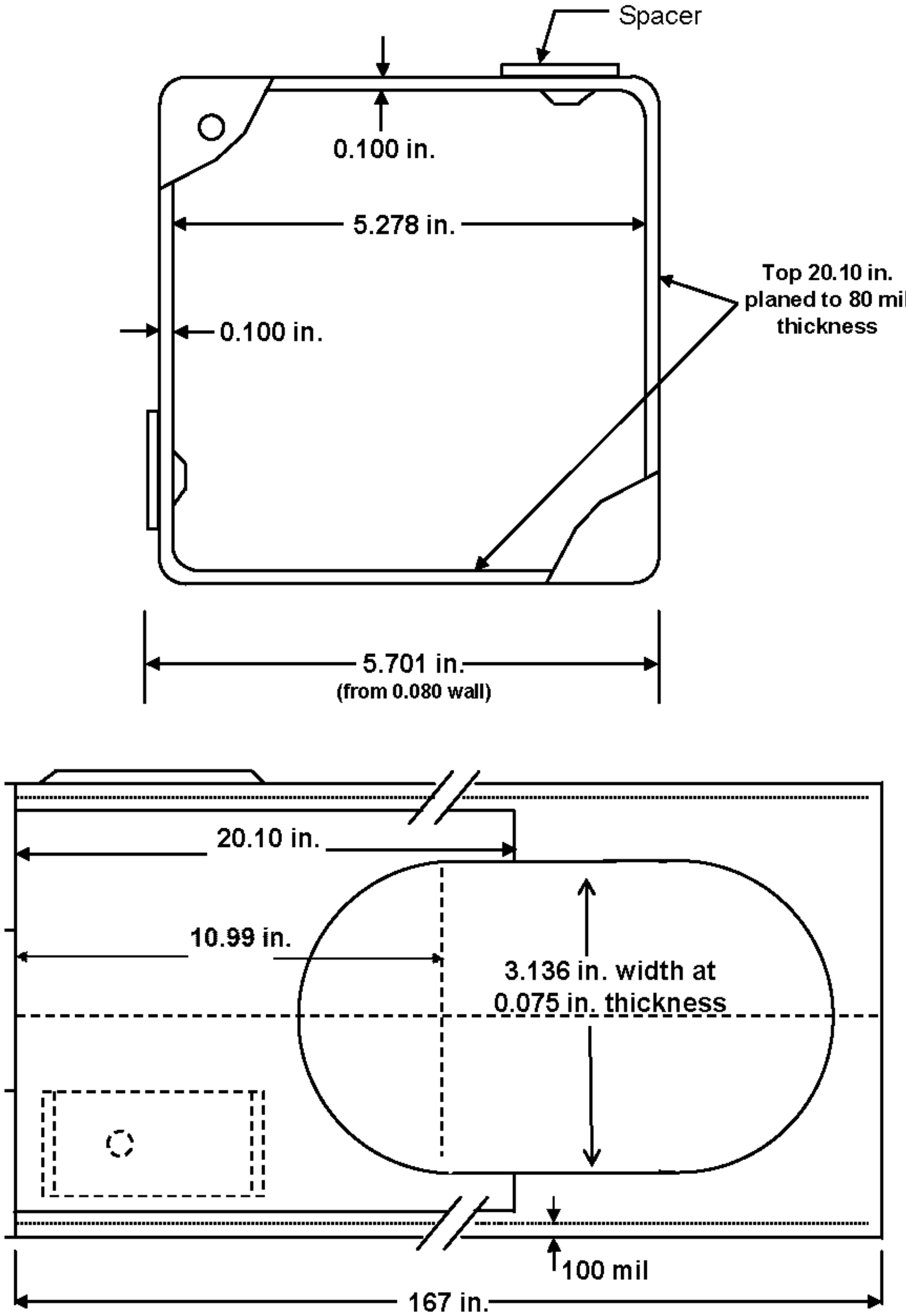
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

100 MIL FUEL CHANNEL

FIGURE 4.2-18-2, Rev. 1

Auto Cad: Figure Fsar 4_2_18_2.dwg



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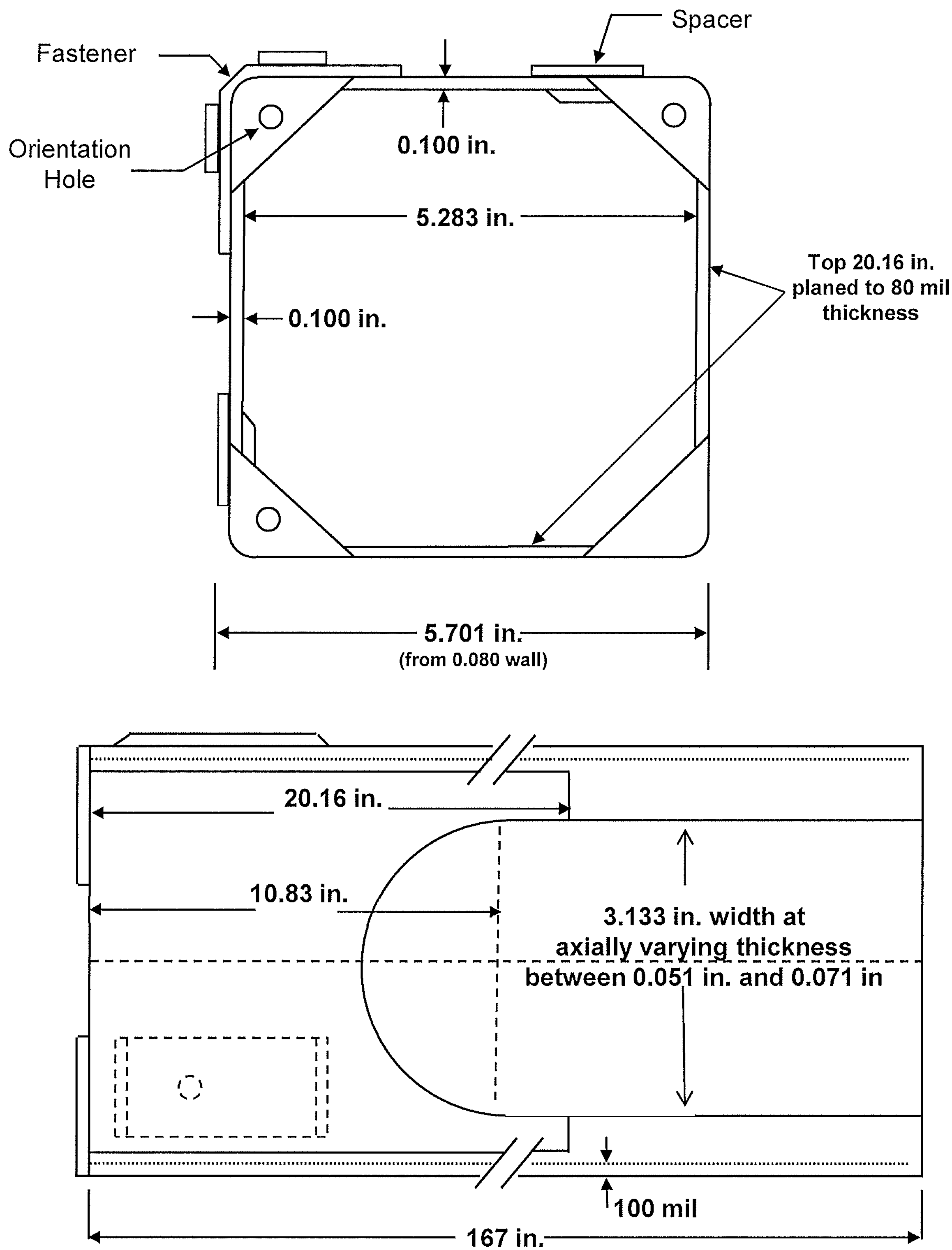
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ADVANCED FUEL CHANNEL

FIGURE 4.2-18-3, Rev. 0

Auto Cad: Figure Fsar 4_2_18_3.dwg



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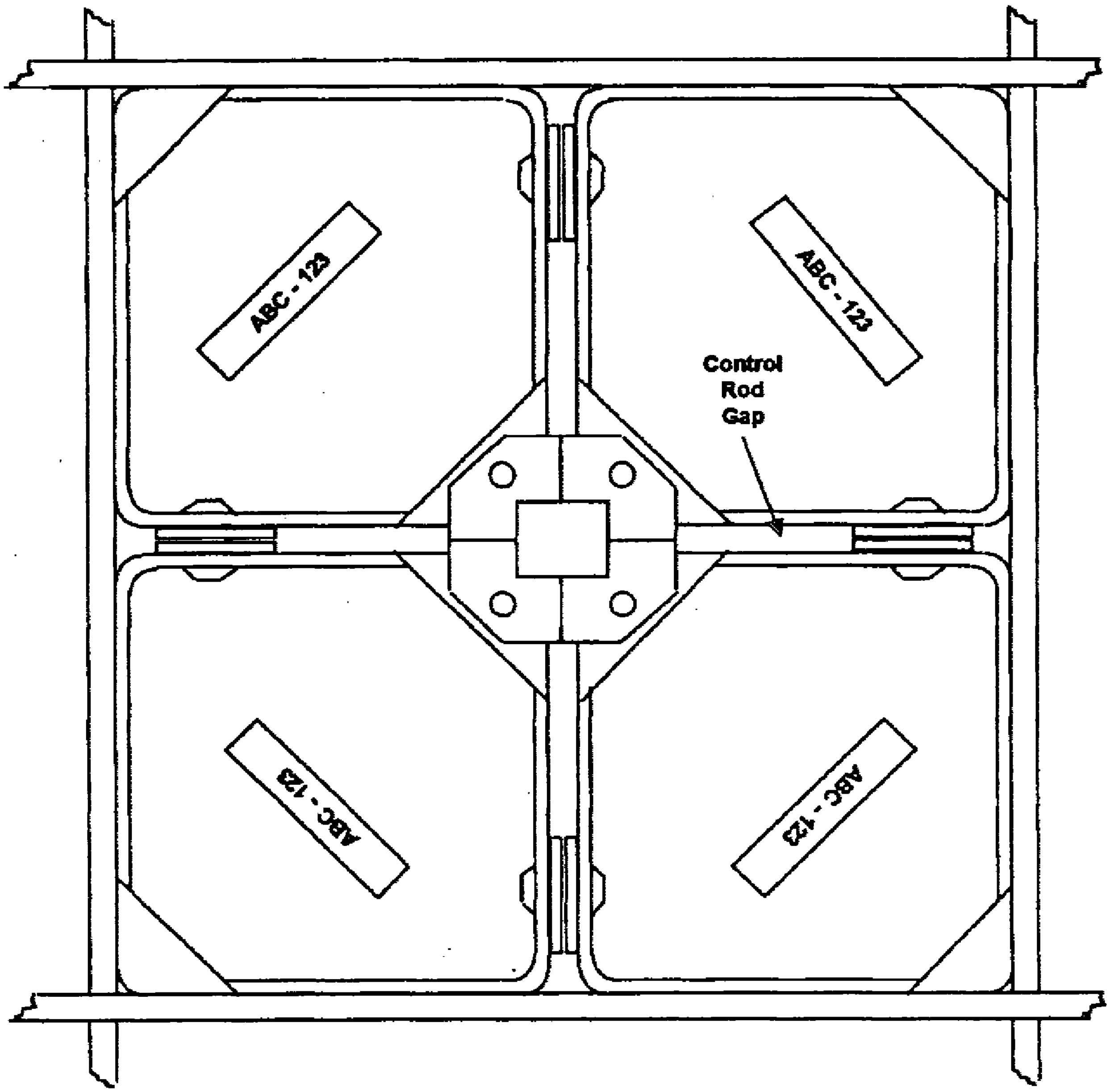
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

ATRIUM 11
ADVANCED FUEL CHANNEL

FSAR FIGURE 4.2-18-4 Rev. 0



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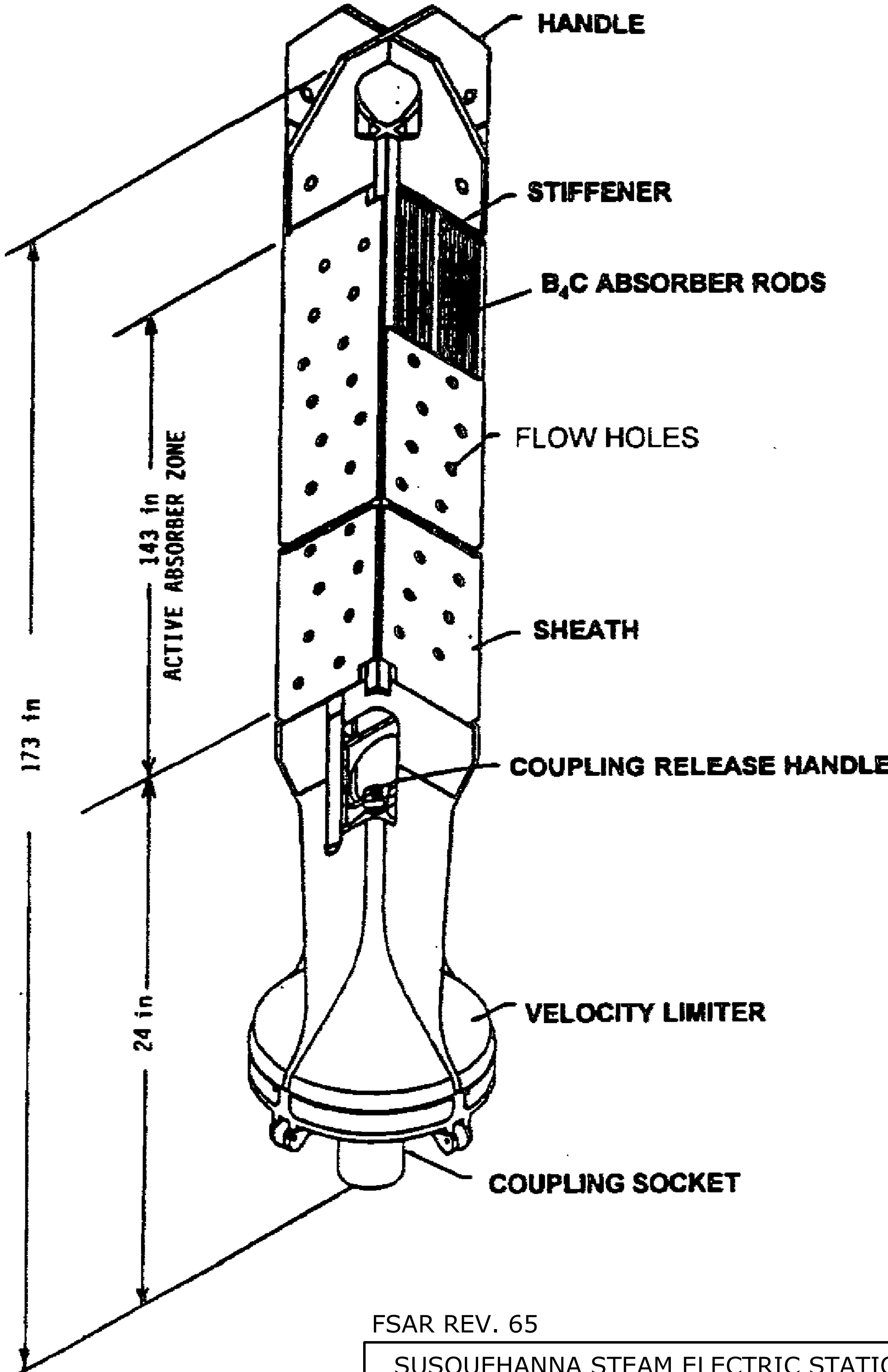
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

CORRECT FUEL ASSEMBLY
ORIENTATION

FIGURE 4.2-19, Rev. 54

Auto Cad: Figure Fsar 4_2_19.dwg



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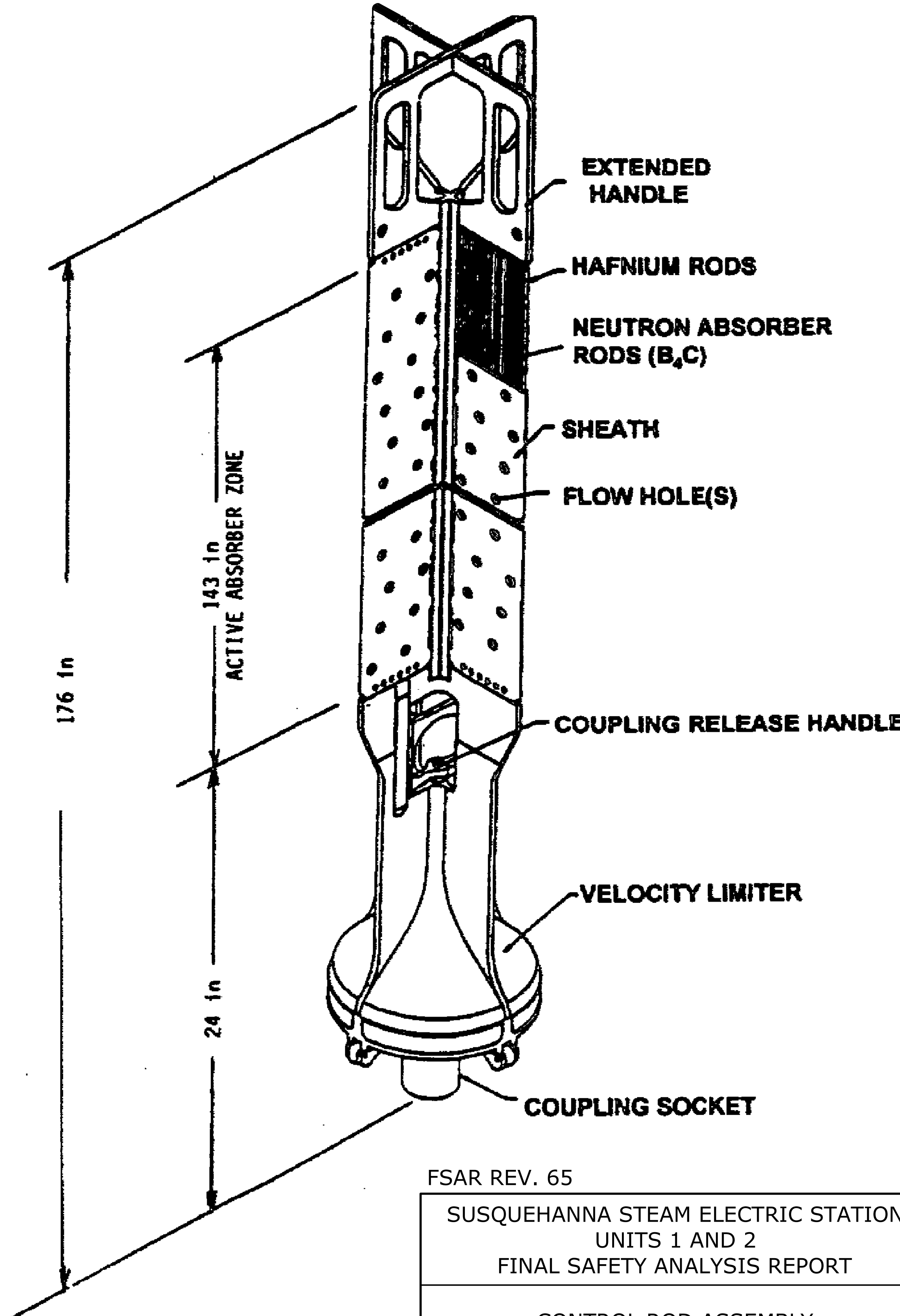
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD ASSEMBLY ORIGINAL EQUIPMENT FIGURE 4.2-20, Rev. 54

Auto Cad: Figure Fsar 4_2_20.dwg



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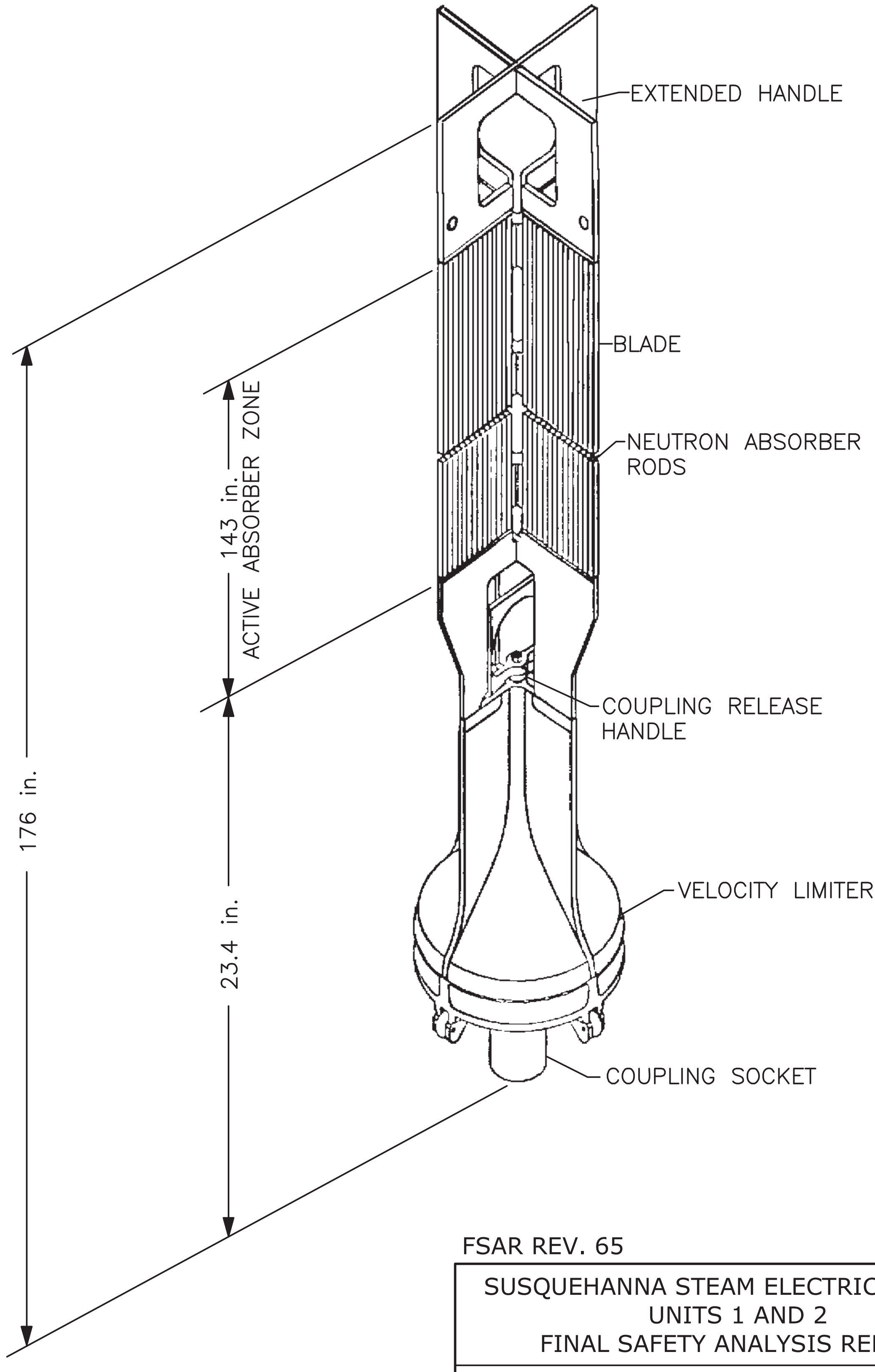
SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD ASSEMBLY DURALIFE 160-C (D-160C) FIGURE 4.2-21, Rev. 54

Auto Cad: Figure Fsar 4_2_21.dwg



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

CONTROL ROD ASSEMBLY
MARATHON

FIGURE 4.2-22, Rev. 2

Auto Cad: Figure Fsar 4_2_22.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
CONTROL ROD ASSEMBLY WESTINGHOUSE CR 99 FIGURE 4.2-23