3.13 COMPLIANCE WITH NRC REGULATORY GUIDES

This section discusses the compliance of the plant design with the guidelines presented in the NRC Regulatory Guides. Where applicable, reference is made to the Final Safety Analysis Report (FSAR) section(s) in which the appropriate design feature is described.

Since the application for the construction permit for this station was docketed in March 1971; therefore, it should be noted that the implementation paragraphs of many of the Regulatory Guides render the provisions contained therein inapplicable to Susquehanna SES by virtue of their effective dates. Nonetheless, the Applicant has evaluated the design and construction against versions of the Regulatory Guides which were current when the application for an Operating License was tendered and has complied with the listed revisions to the extent practicable.

Where compliance to the regulatory guide has been qualified by an interpretation of the regulatory guide, these variances are discussed in either this section or in an appropriate section referenced in a particular response.

The use of an Alternative Source Term (AST) requires changes to source term assumptions and dose acceptance criteria. Regulatory Guides have not been reviewed in detail to determine if exceptions are required for the below listed items. This note captures that the following criteria may be applicable exception(s) to a specific Regulatory Guide.

a. New AST analyses performed in accordance with the guidance in Regulatory Guide 1.183 for the following accidents: Loss of Coolant Accident, Main Steam Line Break, the Refueling Accident and the Control Rod Drop Accident.

b. Dose acceptance criteria is based on the Total Effective Dose Equivalent (TEDE) versus thyroid, whole body and beta dose.

c. Changed from 10CFR100.11 to 10CFR50.67 for dose acceptance criteria.

d. Changed from 10CFR50, Appendix A, General Design Criteria 19 to 10CFR50.67 for control room personnel dose acceptance criteria.

e. No longer committed to Regulatory Guides 1.3, 1.5 and 1.25.

3.13.1 DIVISION 1, REGULATORY GUIDES - POWER REACTORS

Regulatory Guide 1.1 - NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS (November 2, 1970)

As discussed in Subsections 6.2.2 and 6.3.2, Susquehanna SES has been designed to comply with this regulatory guide.

Regulatory Guide 1.2 - THERMAL SHOCK TO REACTOR PRESSURE VESSELS (November 2, 1970)

With respect to the regulatory positions of this regulatory guide, Susquehanna SES is in compliance as follows:

The reactor pressure vessel utilized for Susquehanna SES employs no significant core or vessel design changes from previously approved BWR pressure vessels such as Browns Ferry.

NOTE: Although this regulatory guide has been withdrawn, any prior or existing commitments based on its use are not altered.

An investigation of the structural integrity of boiling water reactor pressure vessels during a design basis accident (DBA) has been conducted (refer to NEDO-10029). It has been determined, based upon methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of a DBA will occur.

The investigation included:

a. a comprehensive thermal analysis considering the effect of blowdown and the low-pressure coolant injection (LPCI) system reflooding;
b. a stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses;
c. the radiation effect on material toughness (NDTT shift and critical stress intensity); and
d. methods for calculating crack tip stress intensity associated with a non-uniform stress field following the design basis accident.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity). Therefore, the results reported in NEDO-10029 provide an upper bound limit on brittle fracture failure mode studies. Because of the upper bound approach, it is concluded that catastrophic failure of the pressure vessel due to the DBA is shown to be impossible from a fracture mechanics point of view. In the case studied, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

For further discussion of fracture toughness of the Reactor Pressure Vessel refer to Subsection 5.2.3.3.1.

Regulatory Guide 1.3 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR BOILING WATER REACTORS (Revision 2, June 1974)

Not Applicable.

Regulatory Guide 1.4 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT FOR PRESSURIZED WATER REACTORS (Revision 2 June, 1974)

Not Applicable.

Regulatory Guide 1.5 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A STEAM LINE BREAK ACCIDENT FOR BOILING WATER REACTORS (March 10, 1971)

Not Applicable.

Regulatory Guide 1.6 - INDEPENDENCE BETWEEN REDUNDANT - STANDBY (ONSITE) POWER SOURCES AND BETWEEN THEIR DISTRIBUTION SYSTEMS (March 10, 1971)

As discussed in Subsections 7.1.2.6, 8.1-6.1.a, and 8.3.1.4, independence between redundant standby (onsite) power sources and between their distribution systems is provided with the exception of Position D.4.c. Swing buses Supply power to the LPCI injection valves and the recirculation piping isolation and bypass valves. Motor generator sets are used to protect redundant power sources from faults that might develop on the swing bus, thus ensuring the requisite degree of independence between the redundant power sources.

Regulatory Guide 1.7 - CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT (Revision 3, March 2007)

The design guidance and assumptions of this regulatory guide are followed as discussed in Subsection 6.2.5.

Regulatory Guide 1.8 - PERSONNEL SELECTION AND TRAINING (Revision 2, April 1987)

Commitment to this regulatory guide is described in FSAR Table 17.2-1 and Technical Specification 5.3.1. Additional information relating to personnel selection and training can be found in FSAR Sections 12.5, 13.1 and 13.2.

Regulatory Guide 1.9 - SELECTION OF DIESEL GENERATOR SET CAPACITY FOR STANDBY POWER SUPPLIES (March 10, 1971, For Diesel Generators 'A-D' and December 1979 for Diesel Generator 'E')

The standby power system is discussed fully in Subsections 8.1.6.1.b and 8.3.1, AC Power Systems. Standby diesel generator power supplies comply with Regulatory Guide 1.9.

Except as indicated below,

(1) Reference: Position C.4 Power quality is in accordance with IEEE 387-1972, Section 5.1.2(4) for Diesel Generators A-D and IEEE 387-1977, Section 5.1.2(5) for Diesel Generator E. At no time during the loading sequence will the frequency or voltage drop to a level which will degrade the performance of any of the loads below their minimum requirements.

(2) Reference: Position C.5. The suitability of each Susquehanna SES Diesel Generator is confirmed by factory qualification testing and preoperational test. Discussion of the factory test results is in Section 8.3.

(3) Reference: Position C.11 (December 1979 Revision only; there is no commensurate requirement in the March 1971 Revision). Exception is taken to the statement that IEEE 387-1977, Section 6.6, "Periodic Testing," should be supplemented by RG 1.108. Regulatory Position C.2.a of RG 1.108 requires that testing of diesel generators occurs at least once every 18 months. The 18 month requirement is interpreted to mean once per refueling cycle. Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. This statement in Regulatory Position C.11 of RG 1.9, and by extension Regulatory Position C.2.a of RG 1.108, clarifies the statement in Section 6.6.2 of IEEE 387- 1977 that operational tests be performed "at acceptable intervals." By taking exception to Regulatory Position C.11 of RG 1.9, exception is also being taken to Regulatory Position C.2.a of RG 1.108 that the frequency of diesel generator testing should be at least once every 18 months. Despite the exceptions to Regulatory Position C.11 of RG 1.9 and C.2.a of RG 1.108, exception is not being taken to the statement in Section 6.6.2 of IEEE 387-1977 that the testing be performed at acceptable intervals. The performance based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

Regulatory Guide 1.10 - MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CATEGORY I CONCRETE STRUCTURES (Revision 1, January 2, 1973)

The testing and inspection program for all mechanical (Cadweld) splices in reinforcing bars of Category I structures are in compliance with this regulatory guide.

Regulatory Guide 1.11 - INSTRUMENT LINES PENETRATING PRIMARY REACTOR CONTAINMENT (March 10, 1971)

The design of the instrument lines penetrating the primary reactor containments of the Susquehanna SES complies with the provisions of this Regulatory Guide. Instrument lines which directly communicate with containment atmosphere, or do not communicate with reactor coolant pressure boundary are treated as extensions of the containment.

Regulatory Guide 1.12 - INSTRUMENTATION FOR EARTHQUAKES (Revision 1 April, 1974)

As described in Subsection 3.7b.4, seismic instrumentation is provided. The instrumentation meets the regulatory position set forth in this guide except for Section C.1.a. since no triaxial peak accelerographs are provided.

Regulatory Guide 1.13 - SPENT FUEL STORAGE FACILITY DESIGN BASIS (Revision 1, December 1975)

The fuel storage facility design basis is described in Section 9.1 and Appendix 9A. Regulatory positions are complied with subject to the following exceptions and clarifications:

(1) Reference: Position C.2. The fuel pool is designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds. The reactor building above the refueling floor consists of steel framing with metal siding.

(2) Reference: Positions C.3 and C.5.a. Interlocks are provided to prevent the 125 ton hook of the reactor building crane from passing over or near stored fuel. These interlocks preclude any load suspended from this crane from tipping over on the stored fuel in the event of a crane failure. The 5 ton auxiliary hook suspended from the same crane trolley is prevented from passing over stored fuel when fuel handling is not in progress by administrative controls. There are no planned transfers of loads heavier than a new fuel element over the stored fuel.

(3) Reference: Position C.8. A Seismic Category I makeup water supply from each emergency service water loop is permanently connected to each spent fuel pool by two independent Seismic Category I piping routes. The make-up is provided for filling the Spent Fuel Pool to the proper level to support operation of the RHR Fuel Pool Cooling mode, and to provide for make-up from evaporative losses during cooling by RHR. The make-up rate is sized based on boiling so as to be conservative. The normal makeup system to the fuel pool is not Seismic Category I.

Regulatory Guide 1.14 - REACTOR COOLANT PUMP FLYWHEEL INTEGRITY (Revision 1, August 1975)

Not Applicable.

Regulatory Guide 1.15 - TESTING OF REINFORCING BARS FOR CATEGORY I CONCRETE STRUCTURES (Revision 1, December 28, 1972)

Testing of reinforcing bars for Category I concrete structures is in compliance with this regulatory guide.

Regulatory Guide 1.16 - REPORTING OF OPERATING INFORMATION-APPENDIX A TECHNICAL SPECIFICATIONS (Revision 4, August 1975)

In lieu of the positions stated in this Regulatory Guide, the reporting of operating information for the Susquehanna SES complies with Technical Specifications, 10CFR50.73 and G.L. 97-02.

Regulatory Guide 1.17 - PROTECTION OF NUCLEAR POWER PLANTS AGAINST INDUSTRIAL SABOTAGE (June 1973)

In lieu of the positions stated in this regulatory guide, the protection of Susquehanna SES against industrial sabotage complies with 10CFR73.

Regulatory Guide 1.18 - STRUCTURAL ACCEPTANCE TEST FOR CONCRETE PRIMARY REACTOR CONTAINMENTS (Revision 1, December 28, 1972)

The compliance with this regulatory guide is achieved subject to certain test modifications as discussed in Subsection 3.8.1.7.1.1.

Regulatory Guide 1.19 - NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS (Revision 1, August 11, 1972)

Nondestructive examination of the primary containment liner welds is conducted as discussed in Subsection 3.8.1.

Regulatory Guide 1.20 - COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING (Revision 2, May 1976)

The vibration assessment program for reactor internals as discussed in Subsections 1.5.1, 3.9.2.4 and NEDE 24057P complies with this regulatory guide.

Regulatory Guide 1.21 - MEASURING, EVALUATING AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (Revision 1, June 1974)

Operation of the radwaste systems will be conducted in accordance with this regulatory guide as permitted by the design. Operation and design of the systems are discussed in Section 11.5.

Regulatory Guide 1.22 - PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTION (February 17, 1972)

As discussed in Subsections 7.1.2.6.4, 7.2.2.1.2.1.2, 7.2.4.1.1.2.2.1, 7.3.2a.1.2.1.3, 7.3.2a.2.2.1.2, 7.3.2a.5.2.1, 7.4.2.1.2.1.3, 7.4.2.2.2.1.3, 7.6.2a.3.2.3.1, 7.6.2a.5.4.2, 8.1.6.1d and 8.3.1.3.1.5 periodic testing of protection system actuation functions complies with this regulatory guide.

Regulatory Guide 1.23 - METEOROLOGICAL MEASUREMENT PROGRAM FOR NUCLEAR POWER PLANTS Second Proposed Revision 1, (April, 1986)

The onsite meteorological system and program comply with this regulatory guide.

The commitment to accuracy criteria for delta temperature is Atomic Energy Commission Safety Guide 23, February 1972 (Regulatory Guide 1.23 Rev. 0).

Regulatory Guide 1.24 - ASSUMPTIONS USED FOR EVALUATION THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER GAS STORAGE TANK FAILURE (March 23, 1972)

Not Applicable.

Regulatory Guide 1.25 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS (March 23, 1972)

Not Applicable.

Regulatory Guide 1.26 - QUALITY GROUP CLASSIFICATION AND STANDARDS (Revision 3, February 1976)

In general the requirements of Regulatory Guide 1.26 are met for the Susquehanna Plant. Some exceptions exist due to the purchase date of the NSSS equipment and design changes in a few systems. These exceptions have been documented in correspondence with NRC. Quality Group classifications are detailed in Tables 3.2-1, 3.2-2, and SSES-FSAR 3.2-3.

Regulatory Guide 1.27 - ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (Revision 2, January 1976)

Subject to the exception indicated below, the design of the ultimate heat sink satisfies the requirements of this regulatory guide.

(1) Reference: Position C.2. Position C.2 states that the ultimate heat sink features which are not required to be designed to withstand the Safe Shutdown Earthquake (SSE) should, nonetheless, be designed and constructed to withstand the Operating Basis Earthquake and waterflow based on severe historical events in the region. The requirements of this regulatory guide without the necessity for any makeup operations following the occurrence of an SSE. Therefore, the potential makeup water sources, such as the cooling tower basins and the makeup system from the Susquehanna River, are not required to perform any safety-related function following the occurrence of any seismic event, and design criteria specified in Position C.2 have not been employed in their design.

Compliance is discussed in Subsection 9.2.7. A description of the analysis performed to demonstrate the ability of the ultimate heat sink to meet the requirements of this Regulatory Guide is presented in Subsection 9.2.7.3.

Regulatory Guide 1.28 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (Revision 1, March 1978)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.29 - SEISMIC DESIGN CLASSIFICATION (Revision 2, February 1976) (Revision 3, September 1978 for the Diesel Generator 'E' Facility)

Subject to the clarifications and/or exceptions indicated below, Susquehanna SES complies with this regulatory guide.

(1) Reference: Position C.1.b. For the NSSS, application of this guide is limited to the reactor core and reactor internals which are engineered safety features.

(2) Reference: Position C.1.d and C.1.g. The normal spent fuel pool cooling system is non-seismic Category I. If a seismic event would occur, cooling of the spent fuel is achieved by use of the RHR Fuel Pool Cooling (RHRFPC) mode as described in Sections 5.4.7.1.1.6, 5.4.7.2.6c, 9.1.3.1, and 9.1.3.3. Either or both of two Seismic Category I ESW makeup water supplies to each pool can provide make-up in support of the RHRFPC mode. Additionally, ESW is capable of supplying make-up for the boiling Spent Fuel Pool (SFP) analysis as described in Appendix 9A.

(3) Reference: Position C.1.e. The Main Steam System (MSS) beyond the outer isolation valves up to and including the turbine stop valves and all branch lines 2 1/2 in. in diameter and larger, up to and including the first valve (including their restraints) are not classified Seismic Category I; because portions of the pipe are routed in a non-Seismic Category I building (the Turbine Building). However, the turbine building has been designed to withstand an SSE as stated in Subsection 3.7b.2.8. Further description of the turbine building is given in Subsection 3.8.4.1; applicable load combinations are given in Table 3.8-10. The subject piping is designed in accordance with ASME Section III, Class 2 requirements for the OBE and SSE as described in Subsection 10.3.3.

(4) Reference: Position C.1.h. The component cooling water portions of the reactor recirculation pumps are not Seismic Class I since they do not involve a safety function.

(5) Reference: Paragraph C.2 of the Regulatory Guide. Items which would otherwise be classified non-Seismic Category I, "but whose failure could reduce the functioning" of items important to safety "to an unacceptable safety level" are to be "designed and constructed so that the SSE would not cause such failure." In addition, Paragraph C.4 of the guide requires that the "pertinent quality assurance requirement of Appendix B to 10 CFR Part 50 should be applied to the safety requirements" of such items. Both of these positions are considered to be adequately met by applying the following practices to such items:

(a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.
(b) Field work is performed under the direction of experienced field construction superintendents and is inspected by the staff of field engineers stationed at the site. The field engineers are responsible for verifying that construction is performed in accordance with the design drawings and specifications and with applicable standard codes and specifications.

(6) Reference: Paragraph C.3 of the Regulatory Guide. Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system requires division of the systems into discrete segments terminated by fixed points, this means that the seismic design cannot be terminated at a seismic restraint, but is extended to the first point in the system which can be treated as an anchor to the plant structure. In addition, Paragraph C.4 of the Regulatory Guide takes the position that "the pertinent quality assurance requirement of Appendix B to 10CFR Part 50 should be applied to the safety requirements" of such items. Both these requirements are considered to be met adequately by applying the following practices to such items:

(a) Design and design control for such items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.

(b) Field audits are performed by representatives of the originating design group to assure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

Regulatory Guide 1.30 - QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT (August 11, 1972)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Program with this guide is described in Table 17.2-1.

Regulatory Guide 1.31 - CONTROL OF STAINLESS STEEL WELDING

Control of stainless steel welding (except NSSS scope of supply) complies with Interim Position on Regulatory Guide 1.31 (Branch Technical Position MTEB 5-1 dated November 24, 1975) except as discussed below.

(1) Reference: Paragraph B.1.b of Interim Position. Austenitic stainless steel welding filler materials used in the fabrication and installation of ASME Section III, Class 1, 2, and 3 components are controlled to deposit from 8 to 25 percent delta ferrite. Welding filler materials 309 and 309L are controlled to deposit from 5 to 15 percent delta ferrite and are used only for welding carbon or low alloy steel to austenitic stainless steel. The use of 309L welding filler material is further limited to the overlay deposit on the carbon or low alloy steel component nozzles or connecting pipe when postweld heat treatment is required.

These limits for delta ferrite in austenitic stainless steel welding materials comply with Interim Regulatory Guide 1.31 because the upper limit of 20 percent delta ferrite does not apply to welds that are not heat treated after welding (Paragraph 3b). Solution heat treatment, although not required after welding, is permitted to avoid sensitization.

The procedure for determining the amount of delta ferrite in each heat or lot of austenitic stainless steel welding material does not comply with the Interim Position of the Regulatory Guide. Determination of delta ferrite is in accordance with ASME Section III, Division 1, 1974 Edition,

Paragraph NB02433, except that an undiluted weld deposit is required for each heat of bare wire used with the Gas Metal-Arc process.

(2) Reference: Paragraph B.2 of the Interim Position. This paragraph is complied with for all tests and examinations required by ASME Section III, Division 1, 1974 Edition.

(3) Reference: Paragraph B.3.a of the Interim Position. This paragraph is not complied with. Magnetic measurement of production welds for delta ferrite is unnecessary when austenitic stainless steel welding materials are controlled to deposit 8 to 25 percent delta ferrite based on chemistry, except for 309 and 309L welding materials which are controlled to deposit 5 to 15 percent delta ferrite based on chemistry.
Three Bechtel projects are committed to measuring production welds for delta ferrite in order to collect data and demonstrate that the welding material controls described above are more than adequate for the purpose of avoiding microfissuring. Since this represents a sufficient number of welds for the purpose of collecting data, measurement of production welds for delta ferrite on this project is not planned.

(4) Reference: Paragraph B.3.b of the Interim Position. This paragraph is complied with for welding material certification.

(5) Reference: Paragraphs B.4.a, b, and c of the Interim Position. These paragraphs are not complied with since measurement of production welds for delta ferrite is not performed.

The NSSS scope of supply control of welding is described in Subsection 5.2.3.4.2.

Regulatory Guide 1.31 - CONTROL OF FERRITE CONTENT IN (DIESEL GENERATOR E ONLY) STAINLESS STEEL WELD METAL (Revision 3, April 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10CFR50, Appendix A, GDC 1.

Regulatory Guide 1.32 - CRITERIA FOR SAFETY RELATED ELECTRIC POWER SYSTEMS FOR NUCLEAR POWER PLANTS (Revision 1, March 1976) for Diesel Generators "A-D" and (Revision 2, February 1977 for the Diesel Generator 'E' Facility)

Subject to the clarifications indicated below, Diesel Generators "A-D" are designed in accordance with Regulatory Guide 1.32, Revision 1 (March 1976). Diesel Generator "E" and the transfer points in the Diesel Generator "A-D" rooms are designed to Regulatory Guide 1.32, Revision 2 (February 1977), as discussed in FSAR Sections 8.1.6.1 and 8.3.2.2.

(1) Reference: Position C.1.c. Exception is taken to the statement that the battery service test be performed at least once every 18 months. Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. The performance-based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

(2) Reference: Position C.1.e. This position is addressed in the response to Regulatory Guide 1.75.

(3) Reference: Position C.1.f. This position is addressed in the response to Regulatory Guide 1.9.

Regulatory Guide 1.33 - QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION) (Revision 2, February 1978)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.34 - CONTROL OF ELECTROSLAG WELD PROPERTIES (December 28, 1972)

The electroslag weld method was not used for the fabrication of any core support structures or any ASME B&PV Section III, Class 1 or 2 vessels and components. Therefore, this regulatory guide is not applicable to Susquehanna SES.

Regulatory Guide 1.35 - IN-SERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENT STRUCTURES (Revision 2, January 1976)

Not Applicable.

Regulatory Guide 1.36 - NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL (February 1973)

As discussed in Subsections 4.5.2.4 and 6.1.1.1, the use of nonmetallic thermal insulation for austenitic stainless steel complies with this regulatory guide.

Regulatory Guide 1.37 - QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments.

Compliance of the operational Quality Assurance program with this guide is discussed in Section 17.2.

Regulatory Guide 1.38 - QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, May 1977)

The Susquehanna SES construction quality program is being conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.39 - HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, September 1977)

Compliance of the Operational Quality Assurance Program is described in Section 17.2.

Regulatory Guide 1.40 - QUALIFICATION TESTS OF CONTINUOUS DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER COOLED NUCLEAR POWER PLANTS (March 16, 1973)

As described in Subsection 3.11.2.2, the present design of Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.41 - PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEM TO VERIFY PROPER LOAD DESIGN ASSIGNMENTS (March 16, 1973)

The requirements of this regulatory guide were met. The testing procedures are outlined in Section 14.2.

Regulatory Guide 1.42 - INTERIM LICENSING POLICY ON AS LOW AS PRACTICABLE FOR GASEOUS RADIOIODINE RELEASES FROM LIGHTWATER-COOLED NUCLEAR POWER REACTORS

Withdrawn March 18, 1976.

Regulatory Guide 1.43 - CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS (May 1973)

This regulatory guide prescribes qualification and production cladding controls for ASME SA 508-2 material made to coarse grain practice. This material is not used for any of the safety class components within the NSSS. ASME SA 508-2 composition material employed on the reactor pressure vessel for Susquehanna SES is produced to fine grain practice.

Regulatory Guide 1.44 - CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL (May 1973)

Subject to the following clarifications and exceptions, the use of unstabilized austenitic stainless steel for components that are part of (a) the reactor coolant pressure boundary, (b) systems required for reactor shutdown, (c) systems required for emergency core cooling, (d) reactor vessel internals required for emergency core cooling, and (e) reactor vessel internals which are relied upon to permit adequate core cooling during any mode of normal operation or postulated accident conditions complies with Regulatory Guide 1.44.

(1) Reference: Position C.1. Contamination of austenitic stainless steel (Type 300 series) by compounds that could cause stress corrosion cracking is avoided during all stages of fabrication and installation. Cleaning is limited to solutions that contain not more than 100 ppm of chlorides. Rinsing or flushing is with water containing less than 100 ppm of chlorides. Foreign substances in contact with austenitic stainless steel (die lubricants, penetrant materials, marking materials, masking tape, etc.) either are controlled to contain the following amounts of contaminates, or are removed immediately following the operation in which they are used.

1. The inorganic halogen content shall be less than 200 ppm by weight.

2. The halogen (inorganic and organic) content shall be less than 1 percent by weight measured in accordance with ASTM D808-63.

3. Sulfur content shall be less than 1 percent by weight as measured in accordance with ASTM D129-64.
4. Total low melting point metal (lead, bismuth, zinc, mercury, antimony, and tin) content shall be less than 200 ppm by weight and no individual metal content shall be greater than 50 ppm by weight.

Completed components are packaged so that they are protected from the weather, dirt, wind, water spray, and other deleterious environmental conditions that may be encountered during shipment and subsequent site storage.

In the field, austenitic stainless steel components are stored clean and dry to prevent contamination. System hydrostatic tests are performed with demineralized water containing less than 10 ppm of chlorides. The influent water quality during final flushing or preoperational testing of the completed system is at least equivalent to the quality of demineralized water as defined in ANSI N45.2.1-1973.

Leachable chlorides and fluorides in nonmetallic insulation materials, which come in contact with austenitic stainless steel, are held to the lowest practical level by the inclusion of the requirements of Regulatory Guide 1.36 in the insulation purchase specifications.

(2) Reference: Position C.2. All grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

(3) Reference: Position C.3. All austenitic stainless steels are furnished in the solution heat treated condition in accordance with the material specification. For material that has been solution heat treated by the material manufacturer, testing to determine susceptibility to intergranular corrosion is performed only when required by the material specification. During fabrication and installation austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800° to 1500°F, except for welding and hot forming. Welding practices are controlled to avoid severe sensitization, as described in (6), and solution heat treatment in accordance with the material specification is required following hot forming in the temperature range 800° to 1500°F. Unless otherwise required by the material specification, the maximum time for cooling from the solution heat treat temperature to below 800°F is 3 minutes. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, or ASTM A 393 may be required if the maximum length of time for cooling to below 800°F is exceeded or the solution heat treat condition is in doubt.

(4) Reference: Position C.4. Use of low carbon (0.03 percent maximum) unstabilized austenitic stainless steel is not required since the reactor coolant meets the conductivity and chloride limits of Table 2 of Regulatory Guide 1.56. However, it is used as described in Section 6(c) below.

(5) Reference: Position C.5. Heat treating austenitic stainless steel in the temperature range 800 to 1500°F is not permitted and solution heat treatment is required following hot forming. Since sensitization is avoided, testing to determine susceptibility to intergranular attack is not performed.

(6) Reference: Position C.6. Welding practices are controlled to avoid severe sensitization in the heat-affected zone of unstabilized austenitic stainless steel as described below. Unless otherwise stated, the position applies to both Bechtel and Bechtel suppliers and subcontractors.

a) Weld Heat Input

Bechtel controls weld heat input during field installation by using shielded metal-arc welding and gas tungsten-arc welding processes only, and by limiting the size of electrodes for each process to 5/32 in. and 1/8 in. diameter maximum, respectively. In addition to these two processes, Bechtel suppliers and subcontractors are permitted to use automatic submerged-arc welding and gas metal-arc welding. Hardsurfacing operations are not included.

b) Interpass Temperature

The interpass temperature is controlled so as not to exceed 350°F.

c) Carbon Content

Susceptibility to sensitization is reduced significantly by selecting materials with the lowest reported carbon content. Specifically Type 304 stainless steel with carbon content limited to .030 maximum or 304L Stainless steel with carbon content limited to .030 maximum was used as follows:

Pipe Description	Size	Material
Head Spray	6"	304SS
Core Spray Influent	12"	304SS
Recirculation System	4"	304SS
Standby Liquid Control	1 1/2"	304LSS
Reactor Water Cleanup (Effluent from Reactor)	4"	304LSS
Instrument Piping	1" & 2"	304LSS
Bottom Drain	4"	304LSS
Vent, Drain, and Test Connections	1"	304LSS

d) Solution Heat Treatment

All austenitic stainless steels are provided in the solution annealed condition. This is accomplished by following the manufacturer solution annealing by water quenching from the solution annealing temperature to below 800°F in 3 minutes. Solution heat treatment is not required after welding.

Severe sensitization is avoided by not permitting heat treatment in the temperature range 800 to 1500°F following welding. This requires a special technique when welding stainless steel safe ends (transition pieces) to carbon or low-alloy steel component nozzles or piping. Specifically, a 309L stainless steel overlay or an Inconel weld overlay is deposited on the component and the component is postweld heat treated. Following final postweld heat treatment of the component, the stainless steel safe end is welded to the weld overlay using 308 or 308L austenitic stainless steel or Inconel type welding materials.

Intergranular corrosion testing is not performed on a routine basis. Performing intergranular corrosion tests for each welding procedure serves no useful purpose when welding practices and reactor coolant water chemistry are controlled as described above.

Regulatory Guide 1.45 - REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS (May 1973)

The design of the leakage detection systems is described in Subsection 5.2.5.1.

Regulatory Guide 1.46 - PROTECTION AGAINST PIPE WHIP INSIDE CONTAINMENT (May 1973)

The criteria given in NRC Branch Technical Position MEB3-1, dated 11-24-75, is used in lieu of the criteria prescribed in Regulatory Guide 1.46 dated May 1973 for the non-NSSS scope of supply. Section 3.6 describes how the present design meets these protection requirements.

This regulatory guide is applicable to the main steam, HPCI, RCIC, RWCU, SLCS and recirculation pipelines within the NSSS scope of supply.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe restraints and compartmentalization was done in consonance with the acknowledgement of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe restraints were engineered to preclude damage based on the pipe break evaluation.

Pipe whip requirements for fluid system piping within the primary containment that under normal operation, has service temperatures 200° F or pressures greater than 275 psig comply with ANS 58.2 - "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture" and Regulatory Guide 1.46 except as delineated in the following criteria for no breaks in Class 1 piping:

1. If Equation 10 of NB-3653-1, ASME Code III results in S_n less than or equal to 2.4 S_m for ferritic or austenitic steels, no other requirement need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

2. If Equation 10 results in S_n between 2.4 S_m and 3,0 S_m for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be less than 0.1.

3. If Equation 10 results in S_n greater than or equal to 3.0 S_m for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not exceed 2.4 S_m and the cumulative usage factor, U, must be less than 0.1.

Regulatory Guide 1.47 - BYPASSED AND INOPERATIONAL STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS (May 1973)

The design, as discussed in Subsections 7.1.2, 7.1.2.6.10, 7.2.2.1.2.1.5), 7.3.2a.1.2.1.7, 7.3.2a.2.2.1.5, 7.3.2a.5.2.5, 7.4.2.1.2.1.7, 7.4.2.2.2.1.71, 7.5.1b.7, 7.5.2a.5.4, 7.5.2b.5, and 8.1.6.1.m (Regulatory Guide 1.47), complies with the provisions set forth in this regulatory guide.

Regulatory Guide 1.48 - DESIGN LIMITS AND LOADING COMBINATIONS FOR SEISMIC CATEGORY I FLUID SYSTEM COMPONENTS (May 1973)

The design loading combinations for non-NSSS systems for Positions C.1 to C.12 are described in Table 3.9-6. Operability of active pumps and valves is assured as described in Subsection 3.9.3.2.

The design limits and loading combinations for seismic category I fluid system components for the Diesel Generator 'E' facility are in compliance with this regulatory guide.

GE practice is representative of industry practice and is in general agreement with the requirements of Regulatory Guide 1.48 with the following clarifications:

a. The probability of an OBE of the magnitude postulated for the Susquehanna SES is consistent with its classification as an Emergency Event. However, for design conservatism, loads due to the OBE vibratory motion have been included under upset conditions. Loads due to the OBE vibratory motion plus associated transients, such as turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the OBE occurrence.
b. The use of increased stress levels for Class 2 components is consistent with industry practice as specified in ASME B&PV Code Section III.

For a comparison of NSSS compliance with Regulatory Guide 1.48 see Table 3.13-1. This comparison reflects a GE practice on BWR 4's and 5's and therefore, is applicable to the Susquehanna SES (see Subsections 3.9.2 and 3.9.3).

Regulatory Guide 1.49 - POWER LEVELS OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

Regulatory Guide 1.49 states, in part,

> Anaylses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level. . . for (a) normal operating conditions, (b) transient conditions anticipated during the life of the facility. . . and (c) accident conditions necessary to evaluate the adequacy of structures, systems and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

Fuel dependent analyses include the effects of a 2 percent power uncertainty factor discussed in Regulatory Guide 1.49. Most of the analyses were performed at 100% power level and the impact of the two percent power uncertainty factor is accounted for either statistically or through the inherent conservatism of the methodology. For three of the analyses, ASME over-pressurization (Section 5.2), loss of feedwater flow (Section 15.2.7), and LOCA-ECCS analyses (Section 6.3.3), the effects of the 2 percent power uncertainty factor are not directly included in methodology used for the analyses. Therefore, these analyses were performed at 102% of CPPU rated power to account for the 2 percent power uncertainty factor. Non-fuel dependent analysis power levels include a two percent power uncertainty factor unless a smaller value is specifically justified or the uncertainty is accounted for in the analysis methods. Additionally, Regulatory Guide 1.49 does not apply to some events that have been historically analyzed from nominal initial conditions, which include the Anticipated Transient Without Scram (section 15.8) and Station Blackout (section 15.9) events.

Thus, the SSES units continue to meet the intent of the Guide, which is to assure that all design calculations are performed at the highest possible power level that the plant can be operating.

Regulatory Guide 1.50 - CONTROL OF PREHEAT TEMPERATURE FOR WELDING OF LOW-ALLOY STEEL (May 1973)

The control of preheat temperature for welding of low-alloy steel is described in Subsection 5.2.3.3.2.1.

Regulatory Guide 1.51 - INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 NUCLEAR POWER PLANT COMPONENTS

Withdrawn July 15, 1975.

Regulatory Guide 1.52 - DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ENGINEERED SAFETY FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT WATER COOLED NUCLEAR POWER PLANTS (Revision 1, July 1976 and Revision 2, March 1978)

The filter adsorber systems are designed to mitigate exposures resulting from a design basis accident. The Control Structure Emergency Outside Air Supply System (CSEOASS) and the Standby Gas Treatment System (SGTS) are the only systems that are subject to the requirements of this regulatory guide.

Subject to the clarifications and/or exceptions indicated below, the general intent of this regulatory guide has been met by the current design of the plant. Items (1) through (10) and (13) apply to Revision 1 and items (11) and (12) apply to Revision 2 of the Regulatory Guide 1.52.

(1) Reference: Position C.2.a. Moisture separators are used only where moisture impingement may be a problem. The SGTS is the only system with moisture separators. Heaters are used on both systems (SGTS and CSEOASS) to control humidity before filtration.

(2) Reference: Position C.2.d. Devices, such as pressure relief valves, are not used on either the CSEOASS or SGTS. Neither filtration system is subject to containment pressures (internal or external) or hazardous pressure surges.

(3) Reference: Position C.2.g. The pertinent pressure drop which is instrumented to signal, alarm, and record in the control room is the pressure drop across the first HEPA filter. The SGTS also alarms on high differential pressure across the entire filter system. The flow rate and low flow alarm also indicate proper functioning of the fan.

(4) Reference: Position C.2.i. Overall design considerations include reduction of radiation exposures during routine maintenance and testing insofar as effectually possible. It is envisioned, however, that workers will not handle filter units after a design basis accident and will thereby avoid exposures associated with immediate post-accident filter handling. Accordingly, no efforts were made toward a unitized atmosphere cleanup train design in the interest of accident exposure reduction.

(5) Reference: Position C.3.d. Since none of the HEPA filters separators are exposed to potential iodine removal spray, the units are not designed for contact with the spray. The referenced military standards (MIL-F-51068C and MIL-F-51079A) have been deleted, but represent acceptable standards for installed (or previously purchased) HEPA filters. New HEPA filters will meet the standards presented in ASME AG-1-1997.

(6) Reference: Position C.3.e through C.3.h. In these sections and all others where reference is made to ORNL-NSIC-65, the reference is understood to be to ERDA 76-21 or ANSI N509 where appropriate.

(7) Reference: Position C.3.i. The adsorber beds are designed for 2.5 mg of iodine (both stable and radioactive) per gram of activated carbon averaged over the bed depth. This is consistent with the background information. Each replacement batch of impregnated, activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, rather than those of Table 2 in Regulatory Guide 1.52, Revision 1, except that the 350 ft^3 batch size limit specified in Table 2 shall be retained.

(8) Reference: Position C.3.k. All systems are designed for low flow in order to control temperature rise. Oxidation effects are considered. A water spray system, provided only to minimize property loss in the event of fire, is designed for the control structure OA supply units, to extinguish a fire by flooding the adsorber units. The fire extinguishing system in the SGTS filter, sprays large quantities of water over the charcoal adsorber, until the charcoal temperature drops below its ignition temperature. The water is removed from the SGTS housing through automatic drain valves. The sprays or quenches are not provided for prevention of fire inception.

(9) Reference: Positions C.4.c and C.4.d. The spacing requirement is applicable to systems requiring operator access to remove filters and adsorber trays. Where unnecessary, the space is not provided, e.g., gasketless carbon absorbers which are filled and emptied externally.

(10) Reference: Position C.4.d. The length of pipe associated with manifolding would promote plate-out of the constituents of the sampled gas stream, thereby resulting in erroneous test results. The test probes are located in readily accessible locations; a minimum run of piping is used and manifolding is not employed.

(11) Reference: Positions C.5.a, C.5.c and C.5.d - In-place testing criteria identified in Paragraph C.5.a, C.5.c and C.5.d of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented, however, the testing frequency for C.5.c is 24 months rather than 18 months.

(12) Reference: Positions C.6.a and C.6.b - Laboratory Testing Criteria for activated carbon identified in Paragraph C.6.a and C.6.b of Rev. 2 dated March, 1978 of the Regulatory Guide 1.52 are implemented with the following exceptions:

a. Representative samples of used activated carbon will be tested at ≤30°C and ≥70% relative humidity and in accordance with ASTM D3803-89.

b. New activated carbon will meet the performance requirements and physical property specifications given in Table 5-1 of ANSI N509-1980.

Note: Table 6.5-2 provides details of how various positions (C1 to C4) of Revision 0 of Regulatory Guide 1.52 are met in the design of SGTS and CSEAOSS.

(13) Reference: Position C.4.e. The frequency and duration of operating the cleanup train with the heater in operation is in accordance with the Plant Technical Specifications and Surveillance Requirements.

Regulatory Guide 1.53 - APPLICATION OF THE SINGLE-FAILURE CRITERION TO NUCLEAR POWER PLANT PROTECTION SYSTEMS (June 1973)

The Susquehanna SES complies with this guide in the design of protection, safeguards actuation, and Class 1E electrical systems. Related considerations pertaining to cable separation and associated circuits are considered in the discussion of Regulatory Guide 1.75 and environmental considerations in the discussion of Regulatory Guide 1.89.

Regulatory Guide 1.54 - QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER COOLED NUCLEAR POWER PLANTS (June 1973)

For the non-NSSS scope of supply, a quality assurance program for coatings was in compliance with this regulatory guide.

For the NSSS scope of supply, the quality assurance records requirements in this regulatory guide were not imposed on painting material and paint application for Susquehanna SES since these coatings cover a relatively small exposed surface area.

Regulatory Guide 1.55 - CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (June 1973)

Concrete placement in Seismic Category I structures is in accordance with the regulatory positions of this guide as described in Appendix 3.8B.

Concrete placement for the Diesel Generator 'El Building is in accordance with ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures" and ANSI N45.2.5 "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

Regulatory Guide 1.56 - MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (June 1973)

GE Report, NEDO-10899, "Chloride Control in BWR Coolants," establishes General Electric's position on water purity.

Regulatory Guide 1.57 - DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (June 1973)

The primary containments for Susquehanna SES are reinforced concrete structures. Nonetheless, the provisions of this regulatory guide are applicable to the following components of each containment:

1) Equipment hatch with personnel lock

2) Equipment hatch

3) Drywell head assembly

4) CRD removal hatch

5) Suppression chamber access hatches

6) Pipe and electrical penetrations

These items were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, for Class MC components, the 1971 Edition with addenda through Summer 1972. Allowable stress limits used for the design are in conformance with Regulatory Guide 1.57, Paragraph C.1.d and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE-3131.2 as specified in the Winter 1973 Addenda. A detailed discussion of these design features is contained in Subsection 3.8.2.

Regulatory Guide 1.58 - QUALIFICATIONS OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL (Revision 1, September 1980)

The Quality Assurance Program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is discussed in Section 17.2.

Regulatory Guide 1.59 - DESIGN BASIS FLOODS FOR NUCLEAR POWER PLANTS (Revision 2 August 1977)

The design basis flood, discussed in Section 2.4, is determined in accordance with the regulatory positions of this guide.

Regulatory Guide 1.60 - DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (Revision 1, December 1973)

The design response spectra used in the analysis of Susquehanna SES, except the Diesel Generator 'E' facility, are different from those of the regulatory guide. A detailed discussion of the design response spectra is presented in Subsection 3.7b.1.

Regulatory Guide 1.61 - DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS (October 1973)

The damping values used in the original seismic design of Susquehanna SES, except the Diesel Generator 'E' facility, are different from the regulatory guide. A detailed discussion of the damping values is presented in Subsection 3.7b.1. For snubber elimination or other piping modifications, damping values from this Regulatory Guide may be used.

Regulatory Guide 1.62 - MANUAL INITIATION OF PROTECTIVE ACTIONS (October 1973)

The provisions for manual initiation of protective actions are described in Subsections 7.2.2.1.2.1.7, 7.2.4.1.1.2.2.4, 7.3.2a.1.2.1.9, 7.3.2a.2.2.1.7, 7.3.2a.5.2.7, 7.4.2.1.2.1.9, 7.4.2.2.2.1.9, 7.6.1b.3.1 and 8.1.6.1.0.

Regulatory Guide 1.63 - ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR WATER COOLED NUCLEAR POWER PLANTS (Revision 1, May 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, the provisions of Revision 1 to this regulatory guide (which supplements IEEE 317-1976) were not specifically considered in the design of Susquehanna SES. The design of the electric penetration assemblies is therefore in compliance with Regulatory Guide 1.63 dated October 1973 (which supplements IEEE 317-1972). Specifically, Sections 4.2.3, 4.2.4, 5.1.6, 5.2.2, 6.2, 6.3.3, and 6.4 of IEEE 317-1976 have not been incorporated.

The penetration assemblies are type tested. There are no provisions for periodic testing under simulated fault conditions.

Electrical penetration circuits are summarized as follows:

1. 480 Volt Circuits

   Loads powered from 480 volt motor control centers are supplied via electrical penetrations equipped with #4 awg, #10 awg, or 4/0 awg copper conductors. Typical single line diagrams for each penetration conductor size are shown on Figure 3.13-1.

   Overcurrent protection for penetrations considers both connected load characteristics and penetration time-current capabilities in accordance with the following guidelines:

   a. 480V motor, less than 1.5 hp (Figure 3.13-2A)

      1. Penetration conductor #10 awg

      2. Overcurrent protection redundant, adjustable, magnetic only circuit breakers with a maximum setpoint of 45 amperes and a minimum setpoint which exceeds 200% of the motor locked rotor current.

b. 480V motor, 10 hp or less (Figure 3.13-2B)

1. Penetration conductor #10 awg

2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 40 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

c. 480V motor, 10.1 to 20 hp (Figure 3.13-3)

1. Penetration conductor #4 awg

2. Overcurrent protection redundant, fixed, thermal magnetic circuit breakers with a maximum thermal trip rating of no more than 100 amperes and a minimum thermal trip rating which exceeds 200% of the motor full load current.

Credit is not taken for penetration protection provided by other overload detectors, such as, motor overloads shown in Figure 3.13-1. Penetration seal withstand curves (Figures 3.13-2, 3.13-3, and 3.13-8) apply for the condition when mechanical seal integrity is maintained, but electrical integrity may be comprised.
Adequacy of the subject molded case circuit breaker selections are demonstrated on Figures 3.13-2, 3.13-3 and 3.13-8 by showing time-current characteristic curves with the protective device total clearing time below, and to the left of the penetration seal withstand curve.

2. <u>120 Volt AC Control Circuits</u>

There are two types of 120 volt AC control circuits to be considered: (1) circuit powered by a control transformer located in an MCC cubicle and (2) circuit powered by a 120 volt AC instrument distribution panel (see Figure 3.13-4). #14 AWG is the minimum size used for control circuits.

a. The motor control circuits are powered by control transformers located in the respective MCC cubicles. Control transformers are sized to meet the requirement of the control circuit (one for each starter). Typically a 120 VA control transformer is used with a NEMA Size 1 starter and a 200 VA is used with a NEMA Size 2 starter. The largest control transformer used in connection with a penetration is 350 VA. The maximum short circuit current that can be delivered by a 350 VA transformer in a control circuit is approximately 100 amps. A fuse, rated 3.2 amp or less, located in the respective MCC provides the circuit protection. Since sustained short circuit current will destroy the control transformer before the integrity of the penetration assembly is compromised, backup fuses are not utilized (refer to Figure 3.13-4).

b. Control circuits that emanate from the fuse control panel will have a 10 amp or lower rated fuse at the panel and as a back-up breaker either an identical fuse in series or a breaker in the 120 V instrument AC distribution panel. 20 amp or smaller breaker or fuse will be used for backup protection (refer Figure 3.13-4). All breakers shown in Figure 3.13-4, Case 2, are molded case and self-actuated (short circuit current trip with manual closing).

3. <u>125 VDC Control Circuits</u>

Each 125 VDC control circuit is protected by a 20 amp or smaller fuse located in a control panel with back-up protection provided by a 20 amp breaker (ITE type E) in the dc distribution panel. The mechanical integrity of the penetration assembly is maintained under overload or faults conditions (refer to Fig. 3.13-5).

4. <u>120 Volt Lighting and Space Heater Circuits</u>

Mechanical integrity of all 120 V lighting and space heater circuits is maintained under overload or fault condition. Each type of circuit is discussed below:

a. Each 120 V lighting circuit is provided with a 20 amp breaker with back-up protection of a 50 amp breaker as shown on Fig. 3.13-6.

b. Each 120 V motor (except reactor recirc pump motor) or motor operated valve (MOV) space heater is provided with a 20 amp or smaller fuse protection. Backup protection is provided by a 20 amp or smaller breaker located in a lighting panel (see 4(a) of above).

c. Each of the 120 V space heater circuits for the reactor recirc pump motors is provided with a 40 amp breaker as the primary protection. Backup protection is provided by a 50 amp breaker located in a 280/120 V lighting panel.

5. <u>Medium Voltage Circuits (above 480 volt)</u>

The only medium voltage equipment located inside the containment are two variable frequency reactor recirc. pump motors. These two recirc. pump motors are fed by two independent 13.8 kV M-G (motor-generator) sets with the generator output rated at 3,920 volts. The M-G sets are located in the turbine building. A calculated M-G set generator decrement curve together with the penetration cable thermal curves is shown on Figure 3.13-7. The maximum short circuit current that can be produced by the generator is 7000 amp asymmetrical. Figure 3.13-7 shows that a line-to-line short circuit (across lines which do not furnish input power to the voltage regulator) of about 6,000 amps can be sustained if the generator field breaker and the main feeder to the M-G set drive motor are not tripped. However, the existing protection schemes have redundant detection devices and redundant protection devices for the penetration circuits. In addition, redundant Class 1E overcurrent protection is provided (one overcurrent relay for each Recirculation Pump Trip (RPT) breaker). This protection scheme ensures that the fault current would be cleared before any damage. Therefore, as shown by Figure 3.13-7, the mechanical integrity of the penetration assembly is maintained under the most severe fault condition.

6. <u>Instrumentation Circuits</u>

The instrumentation circuits are low level signal circuits (ma or mv range) such as thermocouples, RTD, etc. These circuits are current limiting. In addition, instrument cables are not routed with any other types of circuits in the same raceway. Therefore, backup protection is not needed.

The requirement for periodic testing and inspecting of fuses, breakers, and containment circuit protection schemes are stated in SSES Technical Requirements Manual. Requirements for back-up penetration circuit protection are met.

Regulatory Guide 1.64 - QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS (Revision 2, June 1976)

The quality program for the design for the Susquehanna SES is described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this regulatory guide is described in Section 17.2.

Regulatory Guide 1.65 - MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS (October 1973)

Subsections 5.2.4 and 5.3.1.7 describe the materials and inspections for reactor vessel closure studs.

Operation will be conducted in accordance with this regulatory guide with the following exception:

All studs will not be removed from the vessel prior to flooding the reactor cavity. Other means will be employed to prevent corrosion.

Regulatory Guide 1.66 - NONDESTRUCTIVE EXAMINATION OF TUBULAR PRODUCTS (October 1973)

Withdrawn September 28, 1977.

Regulatory Guide 1.67 - INSTALLATION OF OVERPRESSURE PROTECTION DEVICES (October 1973)

This regulatory guide is not applicable to Susquehanna SES since the main steamline safety/relief valves discharge into closed systems.

Regulatory Guide 1.68 - INITIAL TEST PROGRAM FOR WATER-COOLED REACTOR POWER PLANTS (Revision 1, January 1977)

Subject to the clarifications and/or exceptions indicated in Subsection 14.2.7, the provisions of this regulatory guide were met by the test programs instituted during the startup of each unit.

The design of the A-D Diesel Generators meets the intent of this Regulatory Guide. The applicable Appendix A of Regulatory Guide 1.68 provides acceptable preoperational testing criteria for emergency/standby AC power supplies. These testing requirements were implemented and led to meeting the intent of 10CFR50, Appendix B.

Regulatory Guide 1.68 - (TASK SC 704-5) INITIAL TEST (DIESEL GENERATOR E ONLY) PROGRAMS FOR WATER-COOLED NUCLEAR POWER PLANTS (Revision 2, August 1978)

The design of the Diesel Generator "E" meets the intent of this Regulatory Guide. This revision added "Emergency loads supplied should be confirmed to be in agreement with design sizing assumptions used for power supplies" to the applicable Appendix A, Section g(3) of Regulatory Guide 1.68. Methods of control of welding, in fabricating and joining safety-related austenitic stainless steel components and systems, were implemented which led to meeting the intent of 10 CFR 50, Appendix B.

Regulatory Guide 1.68.1 - PREOPERATIONAL AND INITIAL STARTUP TESTING OF FEEDWATER AND CONDENSATE SYSTEMS FOR BOILING WATER REACTOR POWER PLANTS (Revision 1, January 1977)

The preoperational testing and initial startup testing of the feedwater and condensate systems associated with Susquehanna SES were conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.68.2 - INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS (January 1977)

The portion of the initial startup test program used to demonstrate the remote shutdown capability of each unit was conducted in accordance with the provisions of this regulatory guide.

Regulatory Guide 1.69 - CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS (December 1973)

The design and placement of the concrete used for radiation shielding differs from the provisions of this Regulatory Guide. The practices employed for the Susquehanna SES are described in Appendix 3.8B.

Regulatory Guide 1.70 - STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS-LWR EDITION (Revision 2, September 1975)

The format of this FSAR complies with this regulatory guide except that replacement pages will contain a change indicator (vertical bar in the margin of text and table pages) and a page change identification consisting of a revision number. The date of the change will not be shown on replacement pages. This is consistent with the requirements of 10 CFR 50.71(e)(5).

With respect to physical specifications, material submitted may be submitted on CD-ROM in accordance with the guidance of NRC Regulatory Issue Summary 2001-05, dated January 25, 2001.

Regulatory Guide 1.71 - WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY (December 1973)

Exceptions are taken to this regulatory guide as specified below:

(1) Reference: Position C.I. Performance qualifications for field personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX.

For the welder qualifications for the reactor coolant pressure boundary, see Subsection 5.2.3.3.2.3.

Regulatory Guide 1.72 - SPRAY POND PLASTIC PIPING December, 1973

Plastic piping is not used in safety related applications.

Regulatory Guide 1.73 - QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS (January 1974)

The requirements of this regulatory guide are met as described in Section 3.11.

Regulatory Guide 1.74 - QUALITY ASSURANCE TERMS AND DEFINITIONS (February 1974)

The Susquehanna SES construction quality program is described in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.75 - PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS (Revision 1, January 1975)

A. This Regulatory Guide endorses IEEE 384-1974 subject to the additions and clarifications delineated in Section C of the guide. Although there is no requirement for Susquehanna SES to comply with Regulatory Guide 1.75 and IEEE 384, Susquehanna SES follows this separation criteria, subject to the clarifications and exceptions below for the NSSS scope of supply. The paragraphs below reference sections of the IEEE standard in all cases and the specific paragraphs of the regulatory position statement where applicable.

(1) Reference: Regulatory Guide 1.75 and Section 4.5 of IEEE 384-1974

Certain power cables are subject to the same requirements as a Class 1E circuit. This applies to derating, environmental qualification, flame retardance, splicing restrictions, and raceway fill.

(2) Reference: Sections 4.5 and 4.6 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(3) Reference: Sections 5.1.3, 5.1.4, and 5.6.2 of IEEE 384-1974

All annunciator and computer input circuits are classified as non-Class 1E circuits. These non-Class 1E circuits are not separated from Class 1E control circuits within Class 1E panels in which the non-Class 1E circuit derives its input, i.e., circuit breaker auxiliary contact used for computer input, etc., nor are they separated in PGCC cable ducts. These non-Class 1E instrument circuits are considered to be low energy and the probability of these non-Class 1E circuits providing a mechanism of failure to the Class 1E circuits is extremely low.

(4) Reference: Position C.15 of Regulatory Guide 1.75 and Section 5.3.1 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

(5) Reference: Sections 5.6.2 and 5.6.3 of IEEE 384-1974

In general, the circuits for redundant Class 1E systems and the circuits for non-Class 1E systems are located in separate panelboards, boxes, racks, and enclosures. Panels, racks, and boxes that contain wiring and devices for Class 1E circuits are labeled distinctly to externally identify the separation system and grouping. Internal to the enclosures, devices such as relays, switches, and instruments, are uniquely identified. In addition, external cables are color coded and marked to be readily identifiable. These methods of identification are described in Subsection 3.12.3.2.

Where required, physical separation is achieved either by a minimum of 6" horizontal and vertical separation, steel barriers, metallic enclosures, or metallic flexible conduit.

Where the above separation methods are not feasible, one of the separation groups are to be covered with one of the following qualified non-flammable materials:

i. Haveg Industries, Siltemp sleeving type S or woven tape type WT65.

ii. Carborundum, Fiberfrax sleeving type HP144T or woven tape type 3L144T.

These materials have been qualified to be used as separation barriers (Wyle Lab. Test Report No. 56669 dated May, 1980). Applications of these materials are controlled and documented.

(6) Reference: Section 5.5 of IEEE 384-1974

See Section 8.1.6.1 (Regulatory Guide 1.75).

Additional information is found in Sections 7.1, 7.2, 7.3, 7.4 and 7.6

B. Compliance to this Regulatory Guide for the non-NSSS scope of supply is discussed in Section 8.1.6.1 (Regulatory Guide 1.75), and for D/G E in Subsection 8.1.6.1.r.

Regulatory Guide 1.76 - DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS (April 1974)

For all tornado-resistant structures except the Diesel Generator 'E' Building, the following parameters were used in lieu of those presented in Table I of this regulatory guide.

Tangential speed:	300 mph
Translational speed:	60 mph
Rate of pressure drop:	1 psi/sec (for 3 seconds)

A detailed discussion of these parameters is contained in Subsection 3.3.2.

The design basis tornado used for the Diesel Generator 'E' Building is in accordance with this regulatory guide.

Regulatory Guide 1.77 - ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS (May 1974)

Not Applicable.

Regulatory Guide 1.78 - ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (Revision 1 December 2001)

As described in Subsection 6.4.4.2, the steps taken to protect control room habitability conform to requirements of this regulatory guide.

Regulatory Guide 1.79 - PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER REACTORS (Revision 1, September 1975)

Not Applicable.

Regulatory Guide 1.80 - PREOPERATIONAL TESTING OF INSTRUMENT AIR SYSTEMS (June 1974)

The primary containment instrument gas system will be tested in accordance with the requirements of Regulatory Guide 1.80, Sections C1 through C.6. The portions of the instrument air system which supply safety-related equipment will also be tested in accordance with Section C.1 through

C.6 of the Regulatory Guide 1.80 (June, 1974). Loss of air testing will be done in the various system preoperational tests. Systems/components which have separate accumulators will be tested with a loss of air/gas to ensure that the accumulators function in accordance with design. Testing described in Regulatory Guide 1.80, Sections C7 through C10 will not be done in the instrument air system or primary containment gas system tests.

Regulatory Guide 1.81 - SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTI-UNIT NUCLEAR POWER PLANTS (Revision 1, January 1975)

The design of the standby electric power systems which use shared diesels complies with this Regulatory Guide by invoking the provisions of Position C.2. See Section 8.1.6.1.

Regulatory Guide 1.82 - SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS (June 1974)

Not Applicable.

Regulatory Guide 1.83 - IN-SERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES (Revision 1, July 1975)

Not Applicable.

Regulatory Guide 1.84 - CODE CASE ACCEPTABILITY, ASME SECTION III DESIGN AND FABRICATION

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.84 is frequently revised to append new Code Cases. No previous compliance requirements to earlier revisions are negated by updates to this Regulatory Guide.

The Susquehanna SES for the non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case N-316 and N-640 have been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases, a specific request shall be made to the NRC.

ASME Code Case N-516-2 has been approved by the NRC for use during the second 10-year inspection interval at Susquehanna SES in a letter dated February 21, 2002 (Relief Request for Authorization to use Code Case N-516-2 as an Alternative to the ASME Code). The use of Code Case N-516-2 is subject to the following three conditions and limitations:

1. Performance qualifications shall be in accordance with Paragraph 3.2 in Code Case N-516-2, except that immediate retest following a failed mechanical bend test shall be in accordance with ASME, Section IX, QW-320.

2. Procedure qualification shall be in accordance with Paragraph 3.1 in Code Case N-516-2. The Alternative Procedure Qualification Requirements of Paragraph 5.0 shall not be used except as noted in Paragraph 4.(b)(4) for the additional requirements for qualification of filler metal.

3. When welding is to be performed on high neutron fluence Class 1 material, then a mockup, using material with similar fluence levels, should be welded to verify that adequate crack prevention measures were used.

The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class I components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the design and fabrication of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.85 - CODE CASE ACCEPTABILITY, ASME SECTION III MATERIALS

The revision date to this Regulatory Guide is intentionally not stated. Regulatory Guide 1.85 is frequently revised to append new code cases, no previous compliance requirement to earlier revisions are negated by updates to this Regulatory Guide. The Susquehanna SES for non-NSSS scope of supply will use only those code cases listed in this regulatory guide. In accordance with Section D of the Regulatory Guide, Code Cases approved by earlier revisions of the Regulatory Guide may have been invoked. In addition, ASME Code Case 1481-1 has been approved by the NRC for use at Susquehanna Steam Electric Station. Should it be deemed necessary or beneficial to apply other code cases a specific request shall be made to the NRC.
The NSSS scope of supply procedure for meeting the regulatory requirements is to obtain NRC approval for code cases applicable to Class 1 components only. Approval of code cases for Class 2 and 3 equipment was not required at the time of the design of the Susquehanna SES, and is not currently required by 10CFR50.55a. Therefore, GE believes that this procedure in conjunction with 10CFR50APPB and other regulatory requirements provide adequate assurance of quality in the materials of safety-related equipment (see Subsection 5.2.1.2).

Regulatory Guide 1.86 - TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS (June 1974)

The Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.87 - CONSTRUCTION CRITERIA FOR CLASS 1 COMPONENTS IN ELEVATED TEMPERATURE REACTORS (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, 1596) (Revision 1, June 1975)

Not Applicable.

Regulatory Guide 1.88 - COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS (Revision 2, October 1976)

The quality assurance program for the construction of the Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.89 - QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER PLANTS (November 1974)

For the non-NSSS scope of supply, the degree of compliance with this regulatory guide and justification for any exceptions to this guide are provided in Section 3.11.

For the NSSS scope of supply, see Subsections 3.9.2.2a.2.7, 3.9.2.2a.2.9, 3.11.2, 7.2.2.1.2.1.11, and 7.3.2a.2.2.1.10.

Regulatory Guide 1.90 - IN-SERVICE INSPECTION OF PRESTRESSED CONCRETE CONTAINMENT STRUCTURES WITH GROUTED TENDONS (Revision 1, August 1977)

Not Applicable.

Regulatory Guide 1.91 - EVALUATION OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANT SITES (January 1975)

An examination of the historical data for the surface transportation of explosive material near the Susquehanna SES indicates that the commercial truck and rail transportation routes are farther from the station's vital structures than the distances delineated in Figure 2 of this regulatory guide for plants situated in Tornado Region 1, as defined in Regulatory Guide 1.76. Although the closest point of approach from the Susquehanna River to the vital structures of the Susquehanna SES is less than the distance specified in this regulatory guide for the largest probable quantity of explosive material transported by ship (i.e., 5,000 tons of TNT), the Susquehanna River is not commercially navigable for purposes of transporting large quantities of explosives. Therefore, this type of accident is not considered to be probable enough to evaluate.

Regulatory Guide 1.92 - COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS (Revision 1, February 1976)

Since the construction permit for the Susquehanna SES was issued in November 1973, the methods of combining modal responses and spatial components in seismic response analysis, as described in this regulatory guide, were not specifically considered in the original design, except for the Diesel Generator 'E' facility. The methods of design and analysis for structures, components, and piping systems that have been employed are described in Sections 3.7a, 3.7b, and 3.9.

Regulatory Guide 1.92 shall be invoked for the analysis of snubber elimination or other piping modifications whenever Code Case N-411 or Regulatory Guide 1.61 damping values are used.

Regulatory Guide 1.93 - AVAILABILITY OF ELECTRIC POWER SOURCES (December 1974)

Compliance with this guide is discussed in Subsection 8.1.6.1.u.

Regulatory Guide 1.94 - QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS (Revision 1, April 1976)

The quality assurance program for the construction of Susquehanna SES is described in the PSAR, Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.95 - PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE (February 1975)

This regulatory guide was superceded by Revision 1 of Regulatory Guide 178.

Regulatory Guide 1.96 - DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS (Revision 1, June 1976)

Subject to the clarification indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Appendix A, Paragraph 6. The design and inspection of this portion of the leakage control system is in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code. The 100% volumetric inspection of this portion of the system is specifically exempted from the requirement for volumetric inspection by paragraph IWB-1220(a) of Section XI of the ASME Boiler and Pressure Vessel Code (Summer 1975 Addenda).

Note: MSIV-LCS information maintained here for historical purposes. The MSIV-LCS has been deleted. The function is now performed by the Isolated Condenser Treatment Method (Section 6.7), approved by the NRC as an alternative.

Regulatory Guide 1.97 - INSTRUMENTATION FOR LIGHT WATER COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (Revision 2, December 1980)

The accident monitoring instrumentation was designed prior to this Regulatory Guide being issued. The instrumentation for accident monitoring has not been evaluated against Revision 1 to the Regulatory Guide. With the exception of the item discussed below, our position on Revision 2 to the Regulatory Guide is provided in PLA-965, dated November 13, 1981. Compliance is described in more detail in the applicable sections of Chapter 7.

The instrumentation for accident monitoring is not specifically identified on the control panels. The control panel layouts and instrument identification at Susquehanna SES are based on good human factors engineering for the presentation of information to the control room operator. The Susquehanna SES Detailed Control Room Design Review validated our choice of instrument identification scheme. That review indicated that the use of redundant labeling as would have been required to specifically identify accident monitoring equipment would serve to confuse and distract operator performance rather than enhance it.

Subject to the clarifications delineated in PLA-2222, the exception above and the exceptions discussed in FSAR section 7.1.2.6.21, the provisions of this regulatory guide are met.

Regulatory Guide 1.98 - ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A RADIOACTIVE OFFGAS SYSTEM FAILURE IN A BOILING WATER REACTOR (March 1976)

Subject to the clarifications or exceptions indicated below, the assumptions of Regulatory Guide 1.98 are followed in the analyses of the offgas system failure in Subsection 15.7.1.

(1) Reference: Position C.4.a. Dose consequences are expressed in terms of REM TEDE. Dose conversion factors are given in Appendix 15B.

Regulatory Guide 1.99 - RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS (Revision 2, May 1988)

The methods of Regulatory Guide 1.99, Revision 2 are followed in the analyses of reactor pressure vessel fracture toughness in Subsection 5.3.

Regulatory Guide 1.100 - SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS (March 1976)

The implementation paragraph of this regulatory guide states that the requirements of the position statements will only be applied to plants that received construction permits after November 16, 1976. The Construction Permit for Susquehanna SES was issued in November 1973 and therefore the guidelines of this regulatory guide were not utilized in the design of this nuclear power station. However, PP&L conducted a reassessment of the original equipment qualification using the criteria contained in Regulatory Guide 1.100, Rev. 1.

Seismic qualification of the safety related electric equipment (non-NSSS scope of supply) has been conducted in accordance with the IEEE Standard 344-1971. Section 3.10 describes the complete qualification methods and procedures that have been utilized.

The safety-related electric equipment (NSSS scope of supply) meets IEEE 323-1971 and IEEE 344-1971.

The Diesel Generator 'E' facility is in accordance with Regulatory Guide 1.100, Rev. 1 and IEEE Standard 344-1975.

Regulatory Guide 1.101 - EMERGENCY PLANNING FOR NUCLEAR POWER PLANTS (Revision 3, August 1992)

The Susquehanna Emergency Plan complies with the provisions of this regulatory guide.

Regulatory Guide 1.102 - FLOOD PROTECTION FOR NUCLEAR POWER PLANTS (Revision 1, September 1976)

The present design of the Susquehanna SES complies with the provisions of this regulatory guide.

Regulatory Guide 1.103 - POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS (Revision 1, October 1976)

Not Applicable.

Regulatory Guide 1.104 - OVERHEAD CRANE HANDLING SYSTEMS FOR NUCLEAR POWER PLANTS (February 1976)

Subject to the clarifications and exceptions indicated below, the safety related overhead crane handling systems of this station comply with the provisions of this regulatory guide.

(1) Reference: Position C.1.b(2). The nil-ductility transition temperature for the structural steel associated with the cranes was not determined as suggested by this position. Position C.1.b(3) states that a cold proof test represents an acceptable alternative to the requirements of Position C.1.b(2). Accordingly, a cold proof test will be performed in accordance with the general procedures and testing frequency suggested in Position C.4.d except as modified in Item 7 below.

(2) Reference: Position C.3.f. The paragraph states that "The fleet angles between individual sheaves for rope should not exceed 1 1/2 degrees.," however, the fleet angles for the cranes that have been purchased for the Susquehanna SES are 3 degrees 7 minutes.

This position also states that "the pitch diameter of the lead sheave should be 30 times the rope diameter for the 180-degree reverse bend, 26 times the rope diameter for running sheaves and drum,...". The pitch diameter of the running sheaves is 24 times that of the wire rope diameter.

Consistent with established industry standards and in light of the limited number of loading cycles, the present design is acceptable.

(3) Reference: Position C.3.g. This position states that the head block, rope reeving system and load block should be subjected to a load test of 200 percent of the design rated load. No such test has been specified for these components. ANSI Standard B30.2 allows for a 125 percent load test of these components and the purchase orders for these cranes have so specified this type of test.

(4) Reference: Position C.3.j. This position suggests that the designer should provide means within the reeving system located on the head or on the load block combinations to absorb or control the kinetic energy of rotating machinery prior to the incident of two blocking or load hangup. As an alternative to the regulatory position, each crane is provided with dual upper limit switches to preclude the possibility of a "two block" occurrence and an overload switch combined with overcurrent and rate of current rise cutouts to prevent a load hangup.

(5) Reference: Position C.3.p. This paragraph states that provisions should be made for manual operation of the brakes.

This regulatory position has not been incorporated in that there has been no provision made for manual bridge and trolley holding brake operation.

The position also recommends that the trolley and bridge speeds be limited to a maximum speed of 30 fpm for the trolley and 40 fpm for the bridge. The maximum speed for the bridge is actually 50 fpm, but the potential effects of this difference in maximum speed is compensated for by the substantial runway length (approximately 320 ft.) and a stepless type bridge speed control. Administrative controls will be instituted to ensure that a maximum trolley speed of 30 fpm is used when the main hoist is loaded.

(6) Reference: Position C.4.b. This position states that the complete hoisting machinery should be allowed to "two block" during the hoisting test and should also be tested for ability to sustain a load hangup condition. The testing of these conditions was not specified in the purchasing documents for these cranes since the reeving system is not designed for two blocking or load hangup as indicated in Item 4 above, which describes the alternate design that has been incorporated in lieu of the regulatory position C.3.j.

(7) Reference: Position C.4.d. This position states that the cold proof test should be performed at or below the minimum operating temperature of the structural members essential to the structural integrity of the crane and should use a dummy load equal to 1.25 times the maximum working load. It further states that "If it is not feasible to achieve the minimum operating temperature during the test, the dummy load should be increased beyond the design rated load 1.5 percent per °F temperature difference." The minimum operating temperature for the cranes is 60°F and by appropriate scheduling of the cold proof test and adjustment of the HVAC systems, it will be attempted to maintain this temperature. However, in no case will the dummy load be increased above an amount of 125 percent of the rated load. Crane testing in excess of 125 percent of the rated load will adversely affect the safety of the cranes, since any such tests may propagate undetectable material defects and thus increase the probability of crane component failures. Furthermore, such testing would violate the ANSI Standard B30.2 and Title 29CFR Part/1910.179(k). This position also states that cold-proof "test frequency should be approximately 40 months or less ...." The cold proof test will be conducted only once

because the manufacturer recommends against repeatedly overloading the crane. Thereafter, all accessible welds whose failure could cause a critical load drop will be nondestructively examined every 4 years or less. This exception is consistent with Section 2.4 of NUREG 0554.

Regulatory Guide 1.105 - INSTRUMENT SPANS AND SET POINTS (Revision 1, November 1976)

Subject to the clarifications and/or exceptions indicated below, the provisions of this regulatory guide are met by the current plant design.

(1) Reference: Position C.4. The guide requires that instrumentation not exhibit certain mechanical characteristics during the environmental qualification tests performed in accordance with Regulatory Guide 1.89. These tests have not been performed for the Susquehanna SES as discussed in the response to Regulatory Guide 1.89.

(2) Reference: Position C.5. The guide requires that a securing device or equivalent be installed on all instrument set point mechanisms. All set points are provided with multiturn, "screwdriver" adjustments that have "built-in" friction to obviate the effect of vibratory inputs.

Regulatory Guide 1.106 - THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES (Revision 0, November 1975; Revision 1, March 1977)

The requirements of Revision 0, November 1975 of this regulatory guide are met for Susquehanna SES except for the 'E' diesel generator. The requirements of Revision 1, March 1977 of this regulatory guide are met for the 'E' diesel generator. Compliance is discussed in Subsection 8.1.6.1.

Regulatory Guide 1.107 - QUALIFICATIONS FOR CEMENT GROUTING FOR PRESTRESSING TENDONS IN CONTAINMENT STRUCTURES (Revision 1, February 1977

Not Applicable.

Regulatory Guide 1.108 - PERIODIC TESTING OF DIESEL GENERATORS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANT (Revision 1, August 1977) (INCLUDING ERRATA, SEPTEMBER 1977)

Subject to the clarifications and exceptions indicated below, the design of the diesel generators is in compliance with the provisions of this regulatory guide (including errata).

(1) Reference: Position C.1.b(5). Test according to requirements in Sections 14.2 and SSES Technical Specifications.

(2) Reference: Position C.2.a. Exception is taken to the statement that testing of diesel generators occur at least once every 18 months. The 18 month requirement is interpreted to mean once per refueling cycle. Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. This statement in Regulatory Position C.2.a clarifies the statement in Section 6.6.2 of IEEE 387-1977 that operational tests be performed "at acceptable intervals." Despite the exception to Regulatory Position C.2.a of RG 1.108, exception is not being taken to the statement in Section 6.6.2 of IEEE 387-1977 that the testing be performed at acceptable intervals. The performance-based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

Regulatory Guide 1.109 - CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10CFR50, APPENDIX I (Revision 1,October, 1977)

The assumptions of Regulatory Guide 1.109 are followed in the analysis of annual doses to man from routine releases presented in Sections 11.2 and 11.3.

Regulatory Guide 1.110 - COSTS-BENEFIT ANALYSIS FOR RADWASTE SYSTEMS FOR LIGHT WATER COOLED NUCLEAR POWER REACTOR (March 1976)

The requirements of this regulatory guide are met.

Regulatory Guide 1.111 - METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASE FROM LIGHT WATER COOLED REACTORS (Revision 1, July 1977)

The assumptions of Regulatory Guide 1.111 are followed in the analyses of atmospheric dispersion factors presented in Section 2.3 and of deposition rates presented in Section 11.3.

Regulatory Guide 1.112 - CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT WATER COOLED POWER REACTORS (April 1976)

The requirements of Regulatory Guide 1.112 are met.

Regulatory Guide 1.113 - ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I (Revision 1, April 1977)

The requirements of Regulatory Guide 1.113 are met.

Regulatory Guide 1.114 - GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT (Revision 1, November 1976)

PP&L will be in compliance with this regulatory guide.

Regulatory Guide 1.115 - PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES (Revision 1, July 1977)

Since the construction permit for the Susquehanna station was issued in November 1973, the methods of turbine missile protection, as described in this Regulatory Guide, were not specifically considered in the design. The design methods and demonstrative analyses employed for the tangential turbine orientation of Susquehanna SES are described in Section 3.5.

Regulatory Guide 1.116 - QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS (Revision 0-R, May 1977)

The Susquehanna SES quality program for construction of safety related items was conducted in accordance with the program described in PSAR Appendix D and amendments. Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.117 - TORNADO DESIGN CLASSIFICATION (June 1976)

All the structures, systems, and components listed in the appendix to this regulatory guide are protected against the effects of tornadoes except that the spent fuel pool is only designed to prevent significant loss of watertight integrity caused by tornadic winds and missiles generated by these winds.

Regulatory Guide 1.118 - PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS (June 1976) and (June 1978 for the 'E' Diesel Generator only)

For compliance with this regulatory guide refer to Section 8.1.6.1.

Regulatory Guide 1.119 - SURVEILLANCE PROGRAM FOR NEW FUEL ASSEMBLY DESIGNS (June 1976)

Withdrawn June 23,1977.

Regulatory Guide 1.120 - FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS (June 1976)

Regulatory Guide 1.120 was withdrawn on August 15, 2001. Susquehanna SES is not committed to this Regulatory Guide. Refer to the Fire Protection Review Report for the Susquehanna SES Fire Protection Program Commitments.

Regulatory Guide 1.121 - BASES FOR PLUGGING DEGRADED PWR STEAM GENERATOR TUBES (August 1976)

Not Applicable.

Regulatory Guide 1.122 - DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR-SUPPORTED EQUIPMENT OR COMPONENTS (September 1976)

The methods used for developing the floor design response spectra for Susquehanna SES are in compliance with the positions of this regulatory guide except as follows:

1. The frequencies used for the calculation of the response spectra are different and are described in Subsection 3.7b.2.5.

2. The procedure for smoothing the spectra (broadening of peaks) is different and is discussed in Subsection 3.7b.2.9.

The above exceptions are Not Applicable for the Diesel Generator 'E' Building where the methods used for developing floor response spectra are in compliance with Regulatory Guide 1.122, Rev. 1 (February 1978).

Regulatory Guide 1.123 - QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR PLANTS (Revision 1, July 1977)

The Susquehanna SES quality assurance program for the construction phase is detailed in PSAR Appendix D and amendments. Compliance of the operational Quality Assurance Program with this regulatory guide is discussed in Section 17.2.

Regulatory Guide 1.124 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS 1 LINEAR-TYPE COMPONENT SUPPORTS (November 1976)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Section 3.9.

Regulatory Guide 1.125 - PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS (March 1977)

No physical models were used during the design of Susquehanna SES.

Regulatory Guide 1.126 - AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION (Revision 1, March 1978)

This position will be supplied later.

Regulatory Guide 1.127 - INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS (April 1977)

Susquehanna SES will comply with this regulatory guide.

Regulatory Guide 1.128 - INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and October 1978 For the Diesel Generator 'E' Facility)

Installation design and installation of Class 1E batteries are in compliance with this regulatory guide except the design ambient temperature of the battery rooms is 80°F - 5°F and 104°F (Max.) to 65°F (Min.) for the Diesel Generator 'E' facility.

Regulatory Guide 1.129 - MAINTENANCE, TESTING AND REPLACEMENT OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS (April 1977 and February 1978 for the Diesel Generator 'E' Facility)

Subject to the clarifications and/or exceptions indicated below the Susquehanna SES complies with Regulatory Guide 1.129 dated April, 1977 and February, 1978 which invoke IEEE Std. 450-1975. As a result of the conversion from the Current Technical Specification to the Improved Technical Specification, IEEE Std. 450-1995 has been established as the applicable IEEE standard for the Class 1E station battery maintenance, testing and replacement. Compliance with Regulatory Guide 1.129 dated April, 1977 and February, 1978 remains unchanged because the intent of Regulatory Guide 1.129 is not changed as a result of the commitment to IEEE Std. 450-1995.

(1) Reference: Position C.1. Exception is taken to the statement that the battery service test be performed at the frequency described in Regulatory Guide 1.32 (i.e., once every 18 months). Periodic testing is performed in accordance with the Technical Specifications, and the test interval may be replaced with performance-based, risk-informed test intervals. The performance-based, risk-informed test intervals are determined in accordance with Technical Specification 5.5.15 as approved by the NRC and are acceptable.

Regulatory Guide 1.130 - DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS I PLATE AND-SHELL-TYPE COMPONENT SUPPORTS (July 1977)

Since the construction permit for Susquehanna SES was issued in November 1973, this regulatory guide was not specifically considered in the design. The methods used to determine design loading combinations for Susquehanna SES are described in Subsection 5.4.14.

Regulatory Guide 1.131 - QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (August 1977)

The electric cables, field splices, and connections for the non-NSSS scope supply are qualified in accordance with IEEE 383-1974. Exceptions to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for Susquehanna SES. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974 were not used.

(2) Paragraph C.4 - Only one aging data point (121 C) has been applied to the cables used on Susquehanna SES.

(3) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(4) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

The electric cables, field splices, and connections for the NSSS scope, of supply have not been evaluated against this regulatory guide.

The electric cables, field splice, and connections for the Diesel Generator 'E' facility and ties to the transfer points in each of the Diesel Generator A-D bays are qualified in accordance with IEEE 383-1974. Exceptions, to the regulatory positions are as follows:

(1) Paragraph C.2 - The design basis event conditions meet the most severe postulated conditions for the Diesel Generator 'E' facility and Diesel Generator A/D bays. Factors for margin given in Section 6.3.1.5 of IEEE 323-1974, were not used.

(2) Paragraph C.6 - Flame tests were done in accordance with IEEE 383-1974. No tests were performed on aged specimen.

(3) Paragraph C.10 - Gas burner position is in accordance with IEEE 383-1974.

(4) Paragraph C. - The gas and air pressures of Section 2.5.4.4.3 were utilized.

(5) Panel internal wires are not qualified to Regulatory Guide 1.131.

Regulatory Guide 1.137 - FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (Revision 0, January 1978 for Diesels A-D, Revision 1, October 1979 for Diesel E)

The design of diesel generators “A-D” meet the intent of the fuel oil quality and testing requirements in Regulatory Guide 1.137, Revision 0. The design of the “E” diesel generator meets the intent of Regulatory Guide 1.137, Revision 1. The diesel generator fuel oil storage and transfer system is discussed/clarified in FSAR Section 9.5.4 and the Fuel Oil Monitoring Program.

Regulatory Guide 1.144 - AUDITING OF QA PROGRAMS (January 1979)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.145 - ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENT AT NUCLEAR POWER PLANTS (NOVEMBER 1982)

Compliance with the Regulatory Guide and Methodology used for the Atmospheric Diffusion Models is described in Section 2.3.

Regulatory Guide 1.146 - QUALIFICATION OF QA PROGRAM AUDIT PERSONNEL (August 1980)

Compliance of the Operational Quality Assurance Program with this guide is described in Section 17.2.

Regulatory Guide 1.148 - FUNCTIONAL SPECIFICATION OF ACTIVE (DIESEL GENERATOR E ONLY) VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS (March 1981)

The design of the "E" Diesel Generator System meets the intent of this Regulatory Guide, which defines operating requirements for safety related valve assemblies. This regulatory guide endorses the use of ANSI Standard N278.1-1975.

Regulatory Guide 1.163 - PERFORMANCE-BASED CONTAINMENT LEAK-TEST PROGRAM (Revision 0, July 1995)

Leakage rate testing of the primary containment for compliance with 10CFR50, Appendix J, Option B is performed in accordance with the guidance provided in this regulatory guide. This regulatory guide also endorses the methodology for testing presented in NEI 94-01 and ANSI/ANS-56.8-1994.

Regulatory Guide 1.183 - ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS (JULY 2000)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Section 6.0. An Alternative Source Term (AST) assessment was not performed for equipment qualification. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

Reference: Section 3.3, Appendix A. No credit is conservatively taken in the AST analyses for fission product reduction due to the initiation of the drywell sprays.

Reference: Section 3.4, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by in-containment recirculation filter systems.

Reference: Section 3.6, Appendix A. No credit is taken in the AST analyses for reduction of airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above.

Reference: Section 3.8, Appendix A. The primary containment is not routinely purged during power operations.

Reference: Section 5.4, Appendix A. The temperature of the leakage does not exceed 212°F.

Reference: Section 7.0, Appendix A. Primary containment purging as a combustible gas or pressure control measure was analyzed and not deemed to be required. Containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis.

Reference: Sections 5.1 to 5.5, Appendix B. Fuel handling and an equipment handling accidents were evaluated within the Reactor Building outside containment.

Reference: Section 2.0, Appendix C. Postulated accident assumes fuel clad failure and fuel melt.

Reference: Section 3.5, Appendix C. The MSIVs and main steam drain lines do not automatically trip closed following a CRDA. The operators manually scram the reactor and close all MSIVs and drain line valves in the event of a main steam line radiation monitor (MSLRM) high-high radiation alarm in accordance with station procedures.

Regulatory Guide 1.194 - ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL ASSESSMENTS AT NUCLEAR POWER PLANTS (JUNE 2003)

Regulatory positions in this guide are complied with subject to the following exceptions and clarifications:

Reference: Sections 3.2.1 – 3.2.4, 3.2.4.1 – 3.2.4.8. The diffusion models used are based on point-source formulations and ground level releases.

Reference: Sections 3.3.2 – 3.3.4. The Control Room ventilation system has only a single outside air intake.

Reference: Sections 4.1 – 4.4. Source-receptor geometry, type and locations. All ground level releases were determined per the ARCON96 methodology utilizing standard time intervals; no χ/Q correction per wind speed averaging.

Reference: Section 6.0. Plume rise was not considered in the χ/Q determinations.

Reference: Section 7.0. No experimental data was utilized to calculate the χ/Qs.

Regulatory Guide 1.197 - DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS (May 2003)

The following regulatory portions are compiled with:

Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Section C.1 and C.2 and

(ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2.

TABLE 3.13-1

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48: LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48: DESIGN LIMIT	REGULATORY GUIDE PARAGRAPH	SSES: LOADING COMBINATION (F)	SSES: CODE ALLOWABLE STRESSES	ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 1 Vessels	Upset (U)	(NPC or UPC) + 0.5 SSE	NB-3223 2/	1.a	(NPC OR UPC), 0.5 SSE	3.0 Sm (INCLUDES SECONDARY STRESSES)	NB-3223	GE REFLECTS INDUSTRY POSITION
	Emergency (E)	EPC	NB-3224	1.b	EPC,.5 SSE + TRANSIENT	1.8 Sm	NB-3224	
	Faulted (F)	NPC + SSE + DSL	NB-3225	1.c	NPC + SSE + DSL	APP F - SECT 111	NB-3225	
Class 1 PIPING	U	(NPC or UPC) + 0.5 SSE	NB-3654 2/	1.a	(NPC or UPC), 0.5 SSE	3.0 Sm (INCLUDES SECONDARY STRESSES)	NB-3654	GE REFLECTS INDUSTRY POSITION
	E	EPC	NB-3655	1.b	EPC,.5 SSE + TRANSIENT	2.25 Sm	NB-3655	
	F	NPC + SSE + DSL	NB-3656	1.c	NPC + SSE + DSL	3.0 Sm	NB-3656	
Class 1 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	NB-3223 5/ 1/	2.a	(NPC or UPC), 0.5 SSE	1.65 Sm	NB-3223	GE REFLECTS INDUSTRY POSITION
	E	EPC	NB-3224	2.b	EPC, .5 SSE + TRANSIENT	1.8 Sm	NB-3224	
	F	NPC + SSE + DSL	NB-3225	2.c	NPC + SSE + DSL	APP F - SECTION 111	NB-3225	
Class 1 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	NB-3222 5/ 6/ 7/ 8/	4.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3222	4.a	EPC			
	F	NPC + SSE + DSL	NB-3222	4.a	NPC + SSE + DSL			
Class 1 Valves (Inactive) By Analysis	U	(NPC or UPC) + 0.5 SSE	NB-3223 5/ 4/	2.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3224	2.b	EPC			
	F	NPC + SSE + DSL	NB-3225 2/	2.c	NPC + SSE + DSL			
Class 1 Valves (Inactive) Designed by Either Std or Alternative Design Rules	U	(NPC or UPC) + 0.5 SSE	1.1 Pr	3.a	(NPC or UPC), 0.5 SSE	1.1 Pr	NB-3525	GE REFLECTS INDUSTRY POSITION
	E	EPC	1.2 Pr	3.b	EPC, .5 SSE + TRANSIENT	1.2 Pr	NB-3526	
	F	NPC + SSE + DSL	1.5 Pr	3.c	NPC + SSE + DSL	1.5 Pr	NB-3527	
Class 1 Valves (Active) By Analysis	U	(NPC or UPC) 0.5 SSE	NB-3222 5/	4a.1	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3222 6/ 7/	4a.2	EPC			
	F	NPC + SSE + DSL	NB-3222 8/	4a.3	NPC + SSE + DSL			
Class 1 Valves (Active) Designed by Std or Alternative Design Rules	U	(NPC or UPC) 0.5 SSE	1.0 Pr 6/	5a.1	(NPC or UPC), 0.5 SSE	1.0 Pr (a)	NB-3525	GE REFLECTS INDUSTRY POSITION
	E	EPC	1.0 Pr	5a.2	EPC, .5 SSE + TRANSIENT	1.0 Pr	NB-3526	
	F	NPC + SSE + DSL	1.0 Pr	5a.3	NPC + SSE + DSL	1.0 Pr	NB-3527	

TABLE 3.13-1 (Continued)

COMPARISON WITH REGULATORY GUIDE 1.48

COMPONENT	PLANT CONDITION	NRC REGULATORY GUIDE 1.48: LOADING COMBINATION 1/	NRC REGULATORY GUIDE 1.48: DESIGN LIMIT	NRC REGULATORY GUIDE 1.48: REGULATORY GUIDE PARAGRAPH	SSES: LOADING COMBINATION (F)	SSES: CODE ALLOWABLE STRESSES	SSES: ASME SECTION 111 REFERENCE	HOW SSES COMPARES WITH NRC REGULATORY GUIDE 1.48
Class 2 & 3 Vessels (Division 1) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	1.1S) 9/	6a	(NPC or UPC), 0.5 SSE	σ_m 1.15) (c)	CODE CASE 1607	FAULTED CONDITION NRC MORE CONSERVATIVE GE REFLECTS INDUSTRY POSITION
	E	EPC	1.1S)	6b	EPC, .5 SSE + TRANSIENT	)	NC/NB3321 1 0b)	
	F	NPC + SSE + DSL	1.5S)	6c	NPC + SSE + DSL	σ_m 2.08)		
Class 2 Vessels (Divison 2) of Section VIII of the ASME Code	U	(NPC or UPC) + 0.5 SSE	NB-3223) 2/	7a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	NB-3224)	7b	EPC			
	F	NPC + SSE + DSL	NB-3225)	7c	NPC + SSE + DSL			
Class 2 & 3 Piping	U	(NPC or UPC) + 0.5 SSE	NC3611.1(b) (4)) (c) (b) (1))	8a	(NPC or UPC), 0.5 SSE	1.2 Sh	NC/ND3611 3(b)) NC/ND3611 3(c)) (4) (b))(b)	NRC MORE CONSERVATIVE GE REFLECTS INDUSTRY POSITION
	E	EPC	NC3611.1(b) (4))10/ (c) (b) (1))	8a	EPC,.5 SSE + TRANSIENT	1.8 Sh	CODE CASE 1606)	
	F	NPC + SSE + DSL	NC3611.1(b) (4)) (c) (b) (2)	8b	NPC + SSE + DSL	2.4 Sh	NC/ND3611.3(d)) (SEE NOTE (b)))	
Class 2 & 3 Pumps (Inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma_m < 1.1S > \frac{\sigma_m + \sigma_b}{1.5}$	9.a	(NPC or UPC), 0.5 SSE	NOT APPLICABLE	NOT APPLICABLE	NOT APPLICABLE
	E	EPC	$\sigma_m < 1.1S > \frac{\sigma_m + \sigma_b}{1.5}$	9.a	EPC			
	F	NPC + SSE + DSL	$\sigma_m < 1.2S > \frac{\sigma_m + \sigma_b}{1.5}$	9.b	NPC + SSE + DSL			
Class 2 & 3 Pumps (Active)	U	(NPC or UPC) + 0.5 SSE	$\sigma_m < 1.0S > \frac{\sigma_m + \sigma_b}{1.5}$	10.a	(NPC or UPC), 0.5 SSE	σ_m 1.1S)	CODE CASE 1636	GE RELECTS INDUSTRY POSITION
	E	EPC	$\sigma_m < 1.0S > \frac{\sigma_m + \sigma_b}{1.5}$ 11/	10.a	EPC,.5 SSE + TRANSIENT	) (e)) (c)	NC/ND 3423	
	F	NPC + SSE + DSL	$\sigma_m < 1.0S > \frac{\sigma_m + \sigma_b}{1.5}$	10.a	NPC + SSE + DSL	σ_m 1.2S)	(See Note (b))	
Class 2 & 3 Valves (Inactive)	U	(NPC or UPC) + 0.5 SSE	.1 Pr	11.a	(NPC or UPC), 0.5 SSE	σ_m 1.1S (c)	CODE CASE 1636	EQUALLY CONSERVATIVE
	E	EPC	1.1 Pr	11.a	EPC,.5 SSE + TRANSIENT	}(c)	NC/ND 3521	
	F	NPC • SSE + DSL	1.2 Pr	11.b	NPC + SSE + DSL	σ_m 2.0S)	(See Note (b))	
Class 2 & 3 Valves (Active)	U	(NPC or UPC) + 0.5 SSE	1.0 Pr	12.a	(NPC or UPC), 0.5 SSE	σ_m 1.1S)	CODE CASE 1636	EQUALLY CONSERVATIVE (a)
	E	EPC	1.0 Pr.) 11/	12.a	EPC,.5 SSE + TRANSIENT	} (a)	NC/ND 3621	
	F	NPC + SSE + DSL	1.0 Pr)	12.a	NPC + SSE + DSL	σ_m 1.2S } (c)	(See Note (b))	

TABLE 3.13-1 (CONTINUED)

NOTES FOR COMPARISON TABLE

1. Numerical indicators in the regulatory guide portion of the table (1/, 2/, etc.) correspond to footnotes of the Regulatory Guide 1.48.

2. Alphabetical indicators in Susquehanna SES portion of table (or comparative column) correspond to the following:

   a. In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Subsection 3.9.3.2.

   b. Referenced paragraphs of code currently in course of preparation.

   c. The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150% of that allowed for general membrane (except as limited to 2.45 for inactive components under faulted condition). Refer to Subsection 3.9.5.2.

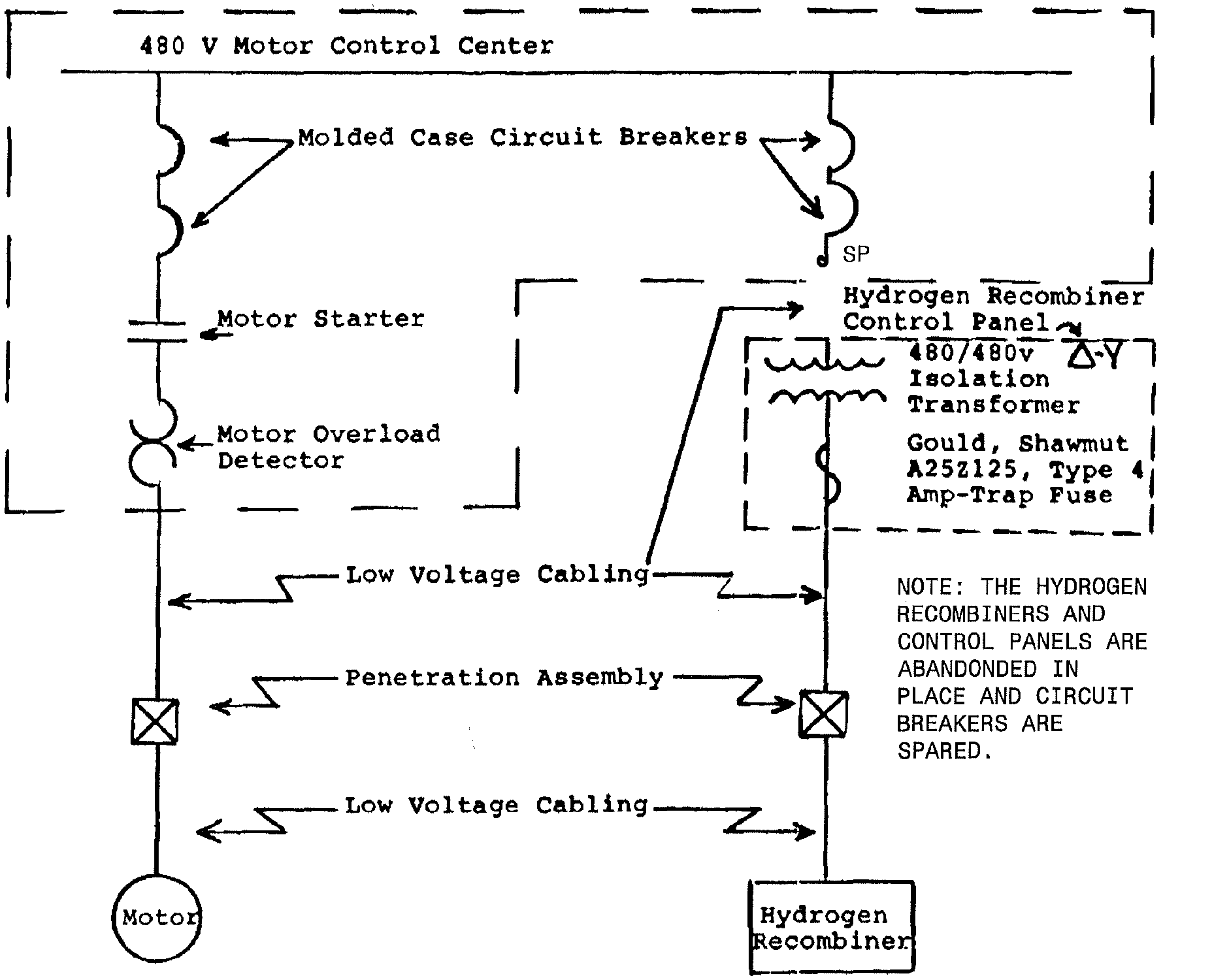
   d. Not used.

   e. Inactive limits may be used since operability will be demonstrated in accordance with Subsection 3.9.3.2.

   f. When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9.2(a).

3. Acronyms

UPC Upset plant conditions
NPC Normal plant conditions
EPC Emergency plant conditions
DSL Dynamic System Loadings
SSE Safe Shutdown Earthquake



FSAR REV.70

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

RELATIONSHIP OF LOADS,
PENETRATION ASSEMBLIES &
PROTECTIVE DEVICES FOR CURVES
PRESENTED IN FIGURES
3.13-2A, 3.13-2B & 3.13-3

FIGURE 3.13-1, Rev. 49

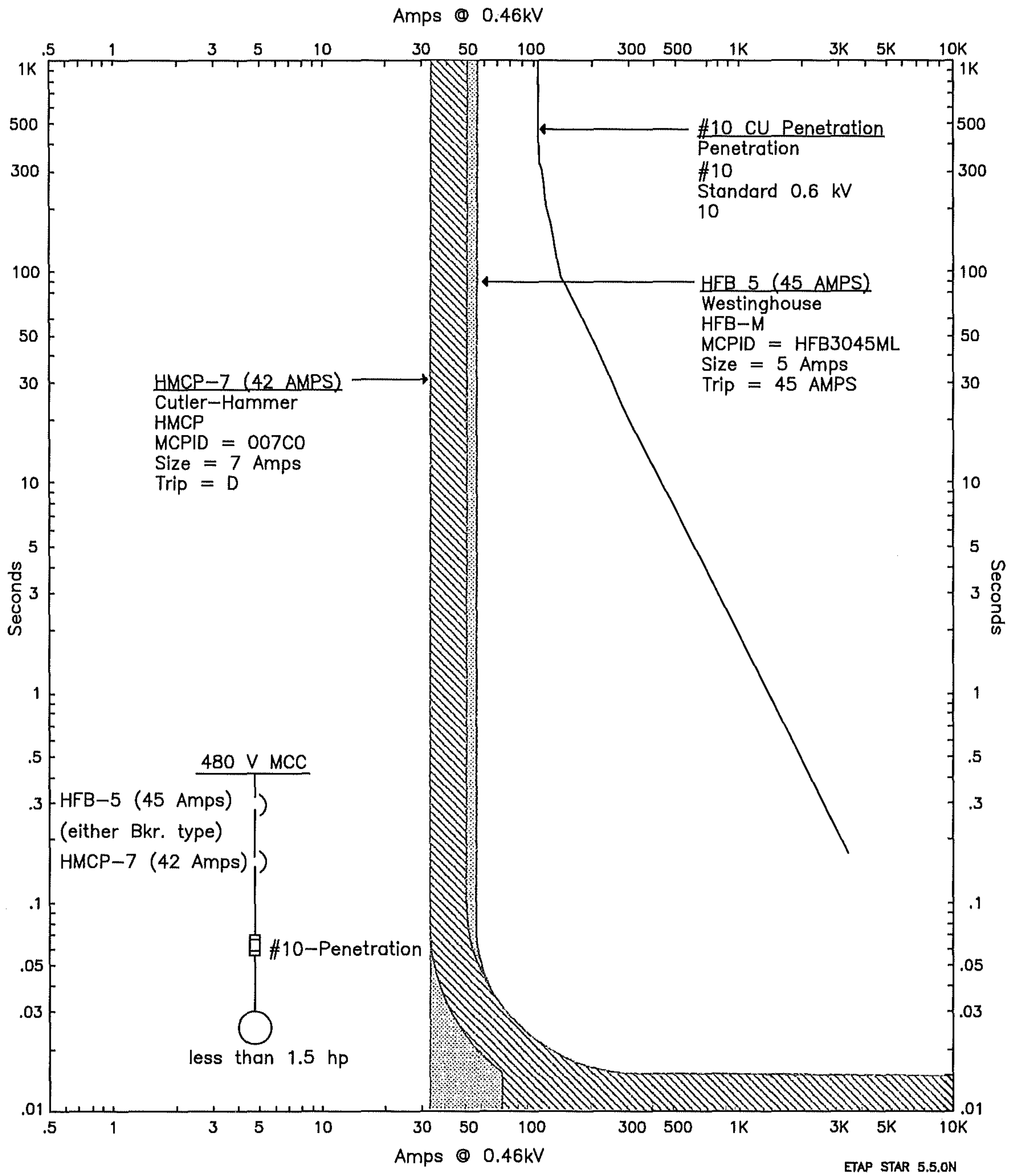
Auto-Cad Figure Fsar 3_13_1.dwg

THIS FIGURE HAS BEEN
REPLACED BY FIGURE 3.13-2A & FIGURE 3.13-2B

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Replaced by FIGURE 3.13-2A & FIGURE 3.13-2B
FIGURE 3.13-2, Rev. 48

AutoCAD; Figure Fsar 3_13_2.doc



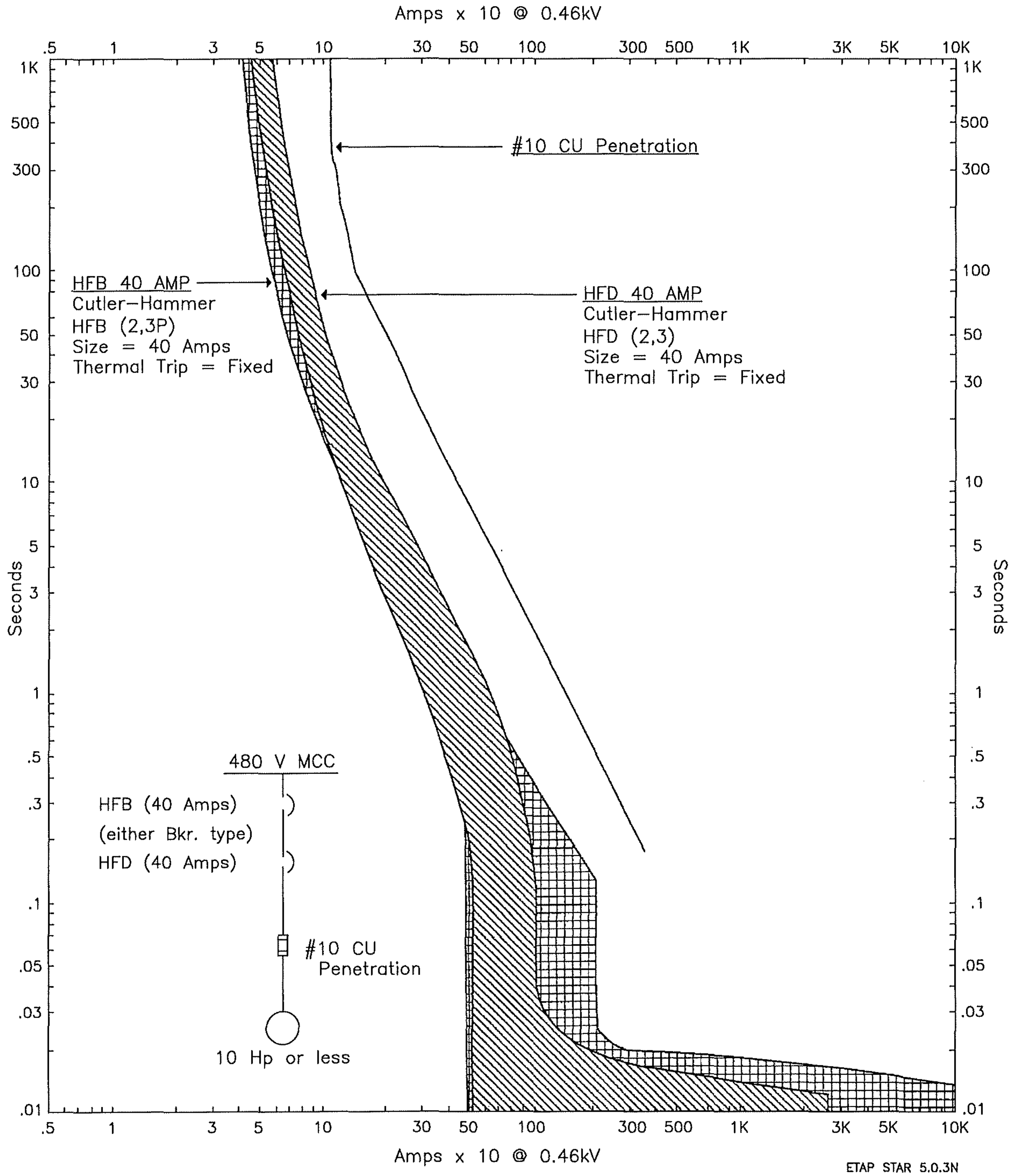
FSAR REV.65

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

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2A, Rev. 1

Auto-Cad Figure Fsar 3_13_2A.dwg



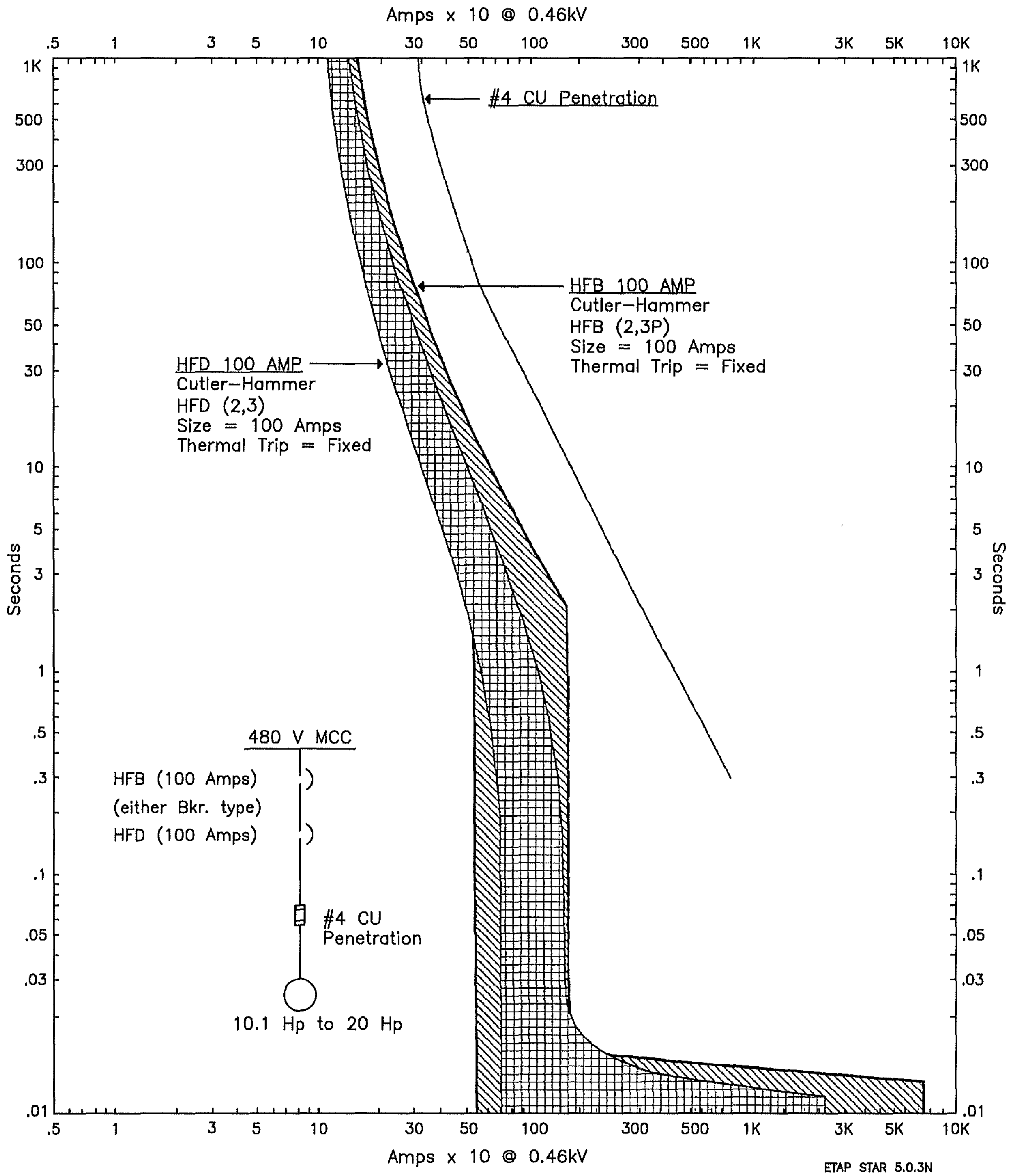
FSAR REV.65

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

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#10 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-2B, Rev. 1

Auto-Cad Figure Fsar 3_13_2B.dwg



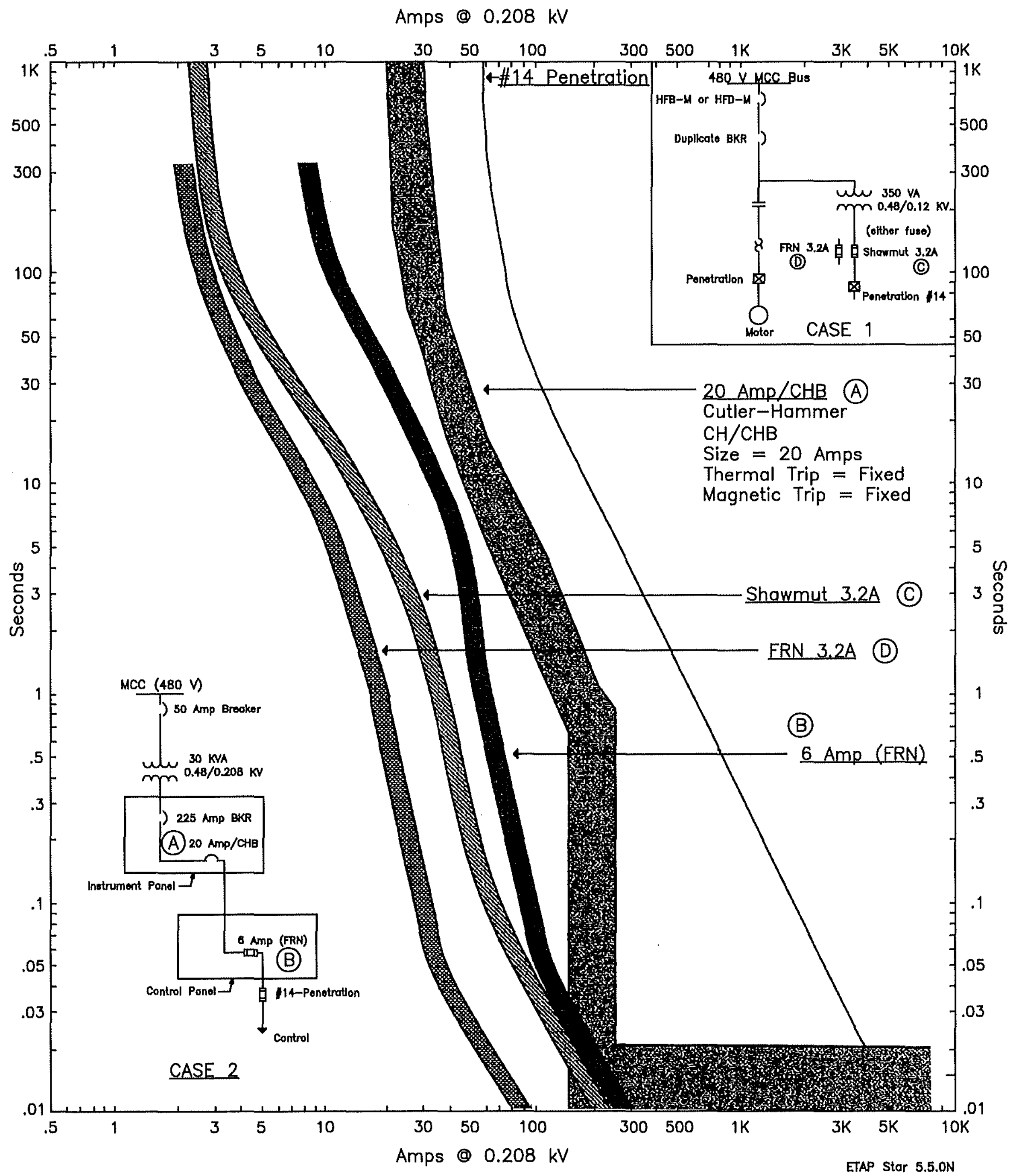
FSAR REV.65

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

TIME-CURRENT
CHARACTERISTIC CURVES FOR
OVERCURRENT PROTECTION OF
#4 COPPER, CONTAINMENT PENETRATIONS

FIGURE 3.13-3, Rev. 48

Auto-Cad Figure Fsar 3_13_3.dwg



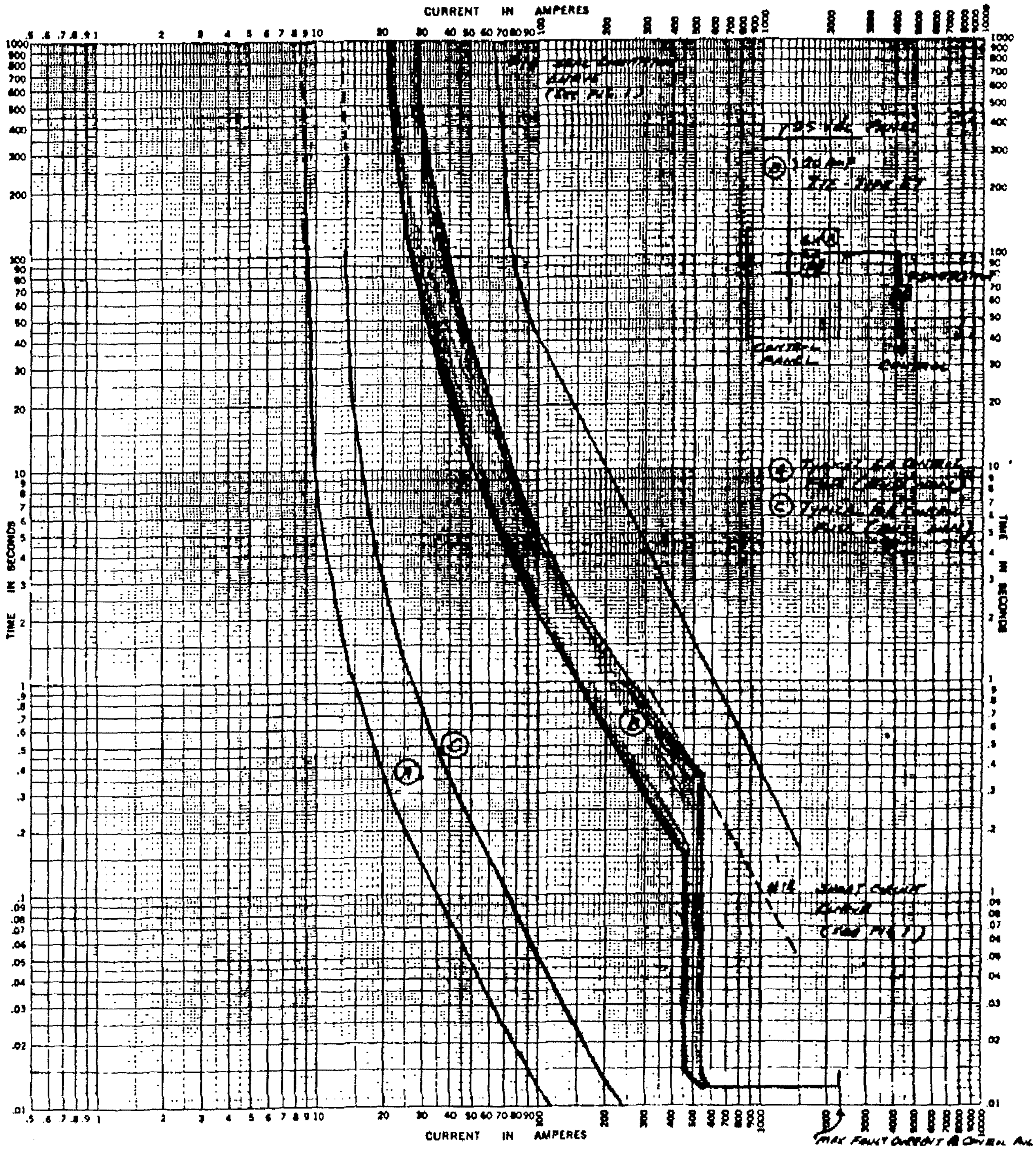
FSAR REV.65

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

120V AC
CONTROL CIRCUITS

FIGURE 3.13-4, Rev. 48

Auto-Cad Figure Fsar 3_13_4.dwg



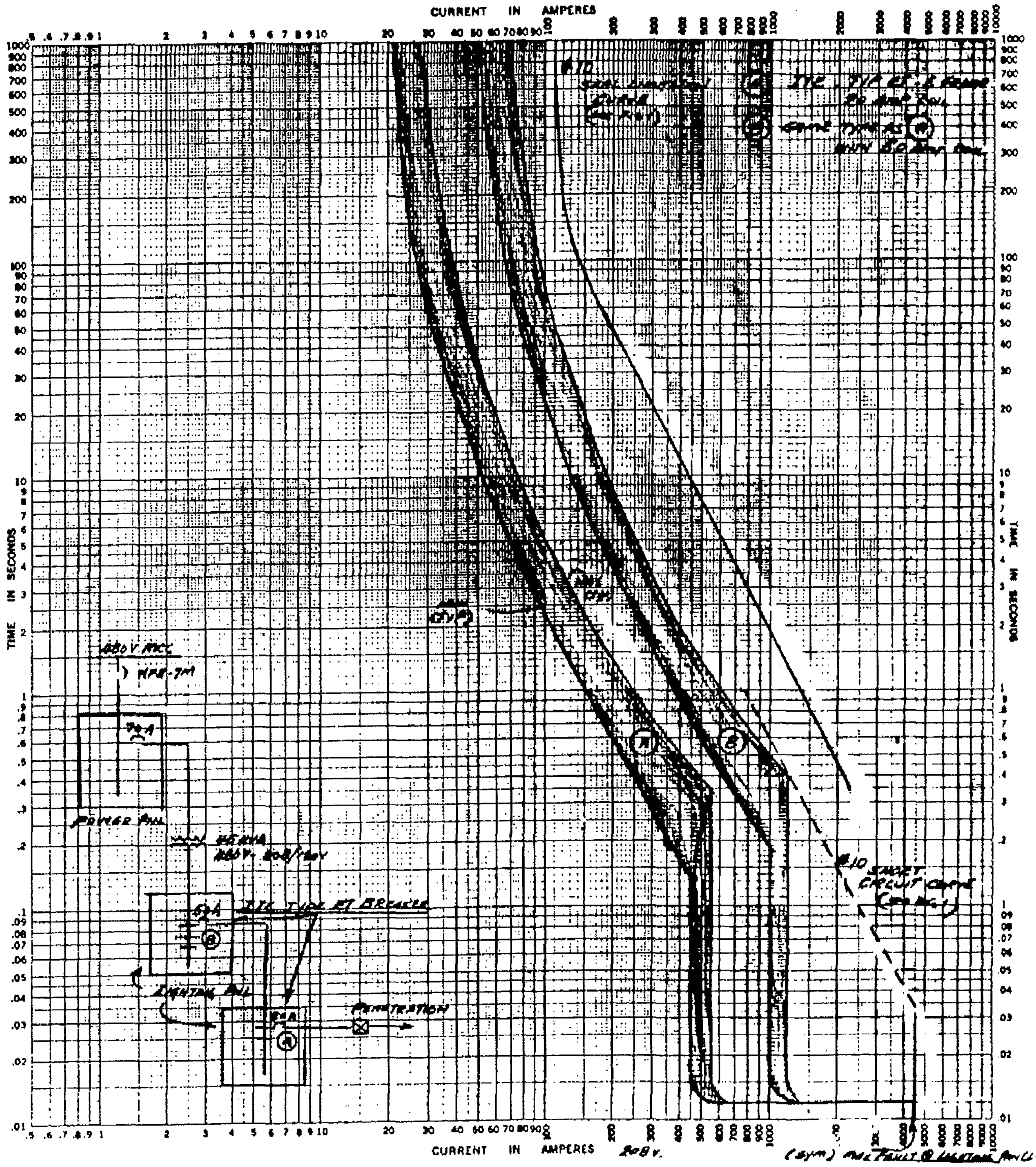
FSAR REV.65

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

125 VOLT DC CONTROL CIRCUITS

FIGURE 3.13-5, Rev. 47

Auto-Cad Figure Fsar 3_13_5.dwg



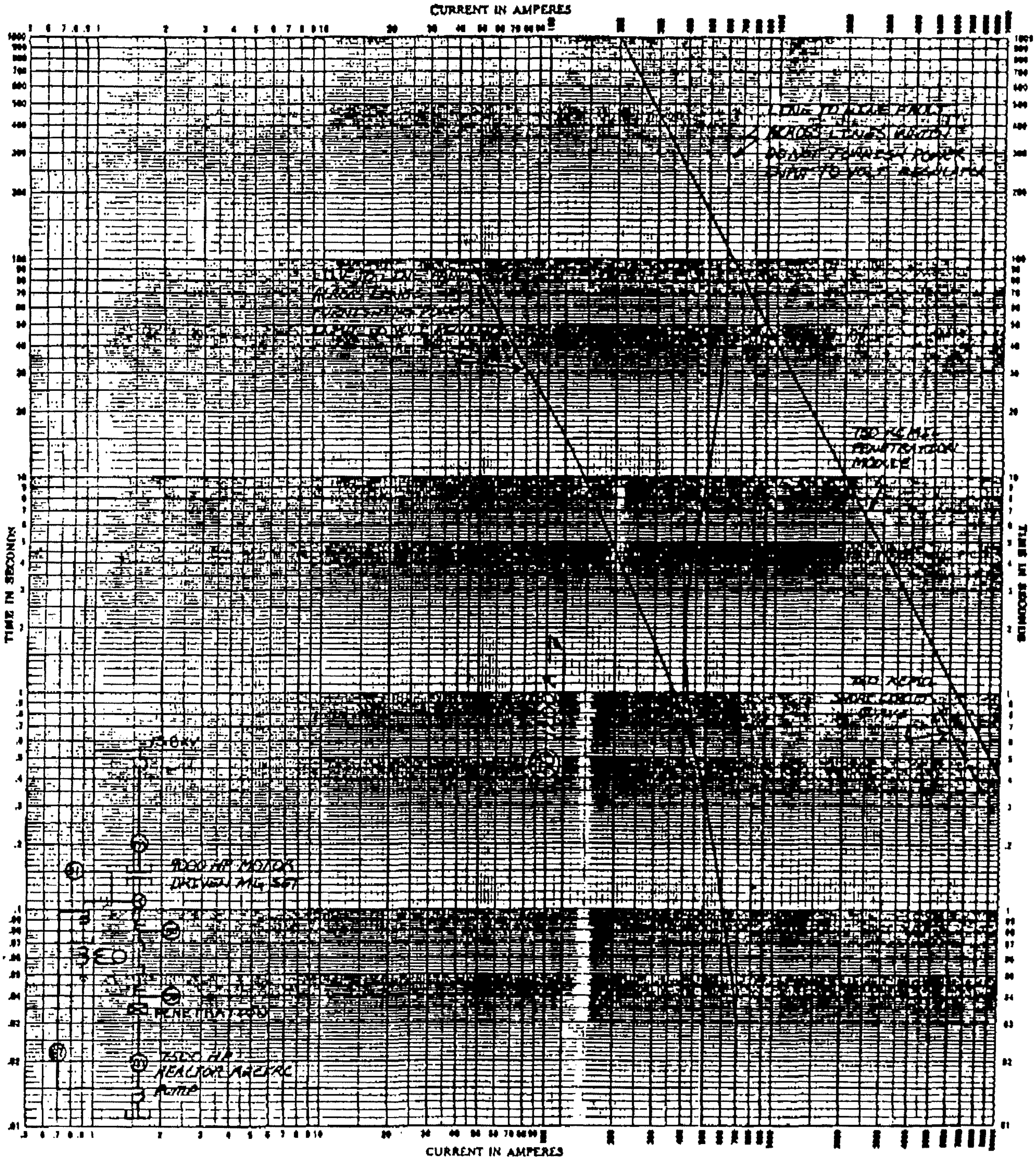
FSAR REV.65

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

208/120 LIGHTING CIRCUITS

FIGURE 3.13-6, Rev. 47

Auto-Cad Figure Fsar 3_13_6.dwg



FSAR REV.65

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

REACTOR RECIRCULATION PUMP-
MAXIMUM SHORT
CIRCUIT CAPABILITY

FIGURE 3.13-7, Rev. 47

Auto-Cad Figure Fsar 3_13_47.dwg