DESIGN EVALUATION SUMMARY
PRELIMINARY REPORT

ISSUED SEPTEMBER 1982

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION - UNIT 3

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A. SAFEGUARDS BUILDING

SECTION 1

AREA SYSTEMS

1.1 CONTAINMENT HEAT REMOVAL SYSTEM (RECIRCULATION SPRAY SYSTEM)

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the recirculation spray (RS) system are:

<u>Standard Review Plans (SRP)</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.2.2, Rev. 3, July 1981, Containment Heat Removal Systems

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Product Cleanup System

<u>Reg Guides</u>

Reg Guide 1.1, Rev. 0, November 1970, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.1.1 Detailed Discussion

The portion of the RS system located within the safeguards area was reviewed to the documents identified above and found to be in basic compliance, although several system modifications would be required to achieve full compliance.

The RS system is designed to the 100 percent mechanical and electrical requirements for an engineered safety features (ESF) system as required by SRP 6.2.2, par. II.1. Confirmation that the postulated single failure (active or passive) will not result in a system loss-of-function would be documented in the system's

Failure Modes and Effects Analyses Report. This report would be prepared per the requirements of SRP 6.2.2, par. II.1.

The piping arrangement of the RS system in the safeguards area was reviewed for failure modes and resulting consequence. It was determined that a passive failure (through wall crack) of the cross connect pipe in its current configuration, which runs between the RS cooler outlet and the decay heat cooler outlet, would subject the RS pumps to the common mode failure of flooding. To ensure RS pump continued operation (prevent loss of function as a result of the single failure), the cross connect line from each RS cooler outlet should penetrate the reactor containment at each cubicle and would be connected by headers in the containment rather than in the safeguards area. This would require four additional 10 in. nominal pipe size containment penetrations and associated containment isolation valves. By this design, each RS pump train would operate independently of the other three in both normal and system accident modes.

As required by SRP 6.2.2, par. II.2 and Reg Guide 1.1, the RS pumps have been designed to ensure adequate net positive suction head (NPSH) under all modes of RS system operation. The calculations which determine NPSH available are consistent with the SRP in that no credit was taken for containment pressure. (Containment pressure equals the vapor pressure of the sump water.)

The requirements of SRP 6.2.2, par. II.3, relative to spray header and nozzle design, are not addressed here since this portion of the system is unique to the reactor containment.

The containment heat removal system for North Anna Unit 3 does not utilize fan coolers as an ESF system. Therefore, the requirements of par. II.4 and par. II.5 of SRP 6.2.2 are not applicable.

SRP 6.3 establishes the design and testing procedures for the emergency core cooling systems (ECCS). The RS system serves as part of the ECCS during the long term following a LOCA and must meet the requirements of SRP 6.3. That portion of the system which performs an ECCS function and is located in the safeguards building should comply with SRP 6.3.

SRP 6.5.2 establishes the design and testing procedures for those containment spray systems that serve as a fission product cleanup system. The RS system meets the design requirements of par. II. 1.a for long-term iodine removed and must meet the requirements of SRP 6.5.2. That portion of the system which performs as a spray function and is located in the safeguards building should comply with SRP 6.5.2.

The RS system is designated as a Class 2 system (Stone & Webster Engineering Corporation (SWEC) safety class) which is consistent with Group B Quality Standards as required by SRP 6.2.2, par. II.6; SRP 3.2.2; and Reg Guide 1.26. The system is also designated Seismic Category I in accordance with the classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 6.2.2, par. II.7.

1.2 SERVICE WATER SYSTEM

Those documents reviewed which provide guidance and/or established criteria for the design basis of the service water system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.2.1, Rev. 1, July 1981, Station Service Water System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.2.1 Detailed Discussion

That portion of the service water system which exists in the safeguards area was reviewed to the documents identified above and found to be in basic compliance. System modifications described in this section would be required to achieve full compliance.

The service water system in the safeguards area cools the containment sump water as it is circulated through the RS coolers. Two redundant supply and return headers are provided for separation of safety flow path. The cross connect which previously interconnected both supply and return headers has been deleted to ensure no single or common mode failure could result in a loss of system function.

The presence of the service water headers in the safeguards area presents the hazard of common mode failure of the RS pump. The passive failure assumption (through wall crack), per SRP 9.2.1, of a single header can result in building flooding. The water, upon reaching the invert elevation of the wall penetrations into the RS pump cubicles, would flow into each cubicle and begin to flood the RS pump lower liner. To mitigate the consequence of the flood, two safety-related, 500 gpm sump pumps would be required at El 238 of the safeguards area. These sump pumps would actuate on rising water level and would discharge to the circulating water discharge tunnel until the break has been located and isolated. Refer to Section A.1.5, Equipment and Floor Drainage System, for a detailed discussion.

The service water supply and return headers and the sump pump and discharge piping are designated Class 3 components (SWEC safety class) which is consistent with Group C Quality Standards as required by SRP 9.2.1, par. II.7; Reg Guide 1.26; and SRP 3.2.2. These portions of the system are also designated Seismic Category I in accordance with the classification requirements of SRP 9.2.1, par. II.8; SRP 3.2.1; and Reg Guide 1.29.

1.3 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

<u>Standard Review Plans</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.5.1, Rev. 2, July 1981, ESF Atmosphere Cleanup Systems

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident ESF Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

1.3.1 Detailed Discussion

The HVAC system for the safeguards building was reviewed to the above documents and the following system modifications would be required to achieve compliance. A discussion of these modifications follows.

The previous design of the HVAC system for the safeguards building consisted of two redundant supply fans located in the safeguards building and two redundant exhaust fans and two common station filter banks located in the auxiliary building. The system provided ESF equipment cooling by distributing outside air within the building and exhausting to the atmosphere.

The current HVAC system in the safeguards building consists of a secondary system used only during normal plant operation and a primary system used exclusively during transient plant conditions.

The secondary HVAC system consists of a nonsafety-related fan, and associated ductwork and dampers. In those areas where this system may adversely impact safety-related systems, the equipment and ductwork will be seismically supported.

The HVAC system used during transient conditions consists of two redundant filter trains with associated fans that discharge to the atmosphere and provide ESF cleanup capability, a redundant source of chilled water to unit coolers which provide ESF equipment cooling capability, associated piping, ductwork, dampers, and instrumentation.

The main reasons for the design changes outlined above are:

1. Reg Guide 1.52 position C.2.f requires that the volumetric flow rate through a single cleanup train be limited to 30,000 cfm or below.

2. VEPCO has requested that maximum design outside air temperatures be increased to 107 FDB and 82 FWB.

With the higher temperature, the previous design (once-through ventilation) would require greater volumetric air flow in order to maintain an adequate ESF equipment ambient temperature. With a greater air flow, unacceptably large systems would result and multiple filters would be required to maintain flow through each filter train at below 30,000 cfm.

SRP 6.5.1 requires additional instrumentation for surveillance of the filter train operation.

1.4 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.4.5, Rev. 3, July 1981, Engineered Safety Feature Ventilation System

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979

1.4.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design, the fire protection system in the safeguards building consisted of yard hydrants located near the building.

To comply with the current documents, a standpipe system with hose stations and water spray systems for the charcoal filter assemblies would be required.

1.5 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 1, September 1974, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 1, August 1973, Seismic Design Classification

1.5.1 Detailed Discussion

The portion of the equipment and floor drainage system (EFDS), which exists in the safeguards area was reviewed to the above documents and systems modifications described in this section would be required to achieve full compliance.

The impact of SRP 6.2.2, concerning passive failure analysis of the RS system, resulted in significant redesign of the safeguards area (refer to Section I.1.1.3). The aerated drain system, which makes up the EFDS in the safeguards building, is not compatible with this redesign. Therefore, as a result of SRP 6.2.2, the following modifications are required to achieve compliance with SRP 9.3.3:

1. A groundwater pump, approximately 12-20 gpm and sized for maximum groundwater inleakage, would be located in each of the four RS pump cubicles. These pumps would discharge directly into the liquid waste system.

2. A sump and a 500 gpm drain pump, sized to mitigate the consequences of a passive service water line failure, would be located within each of the two outer areas in the safeguards building. The pumps would discharge to the circulating water discharge tunnel.

SRP 9.3.3, par. II.2, states that the EFDS should be safety-related if failure or malfunction of a portion of the system could result in adverse effects on essential systems or components (i.e., necessary for safe shutdown, accident prevention, or accident mitigation). Therefore, the aerated drain system within the safeguards building is designated Class 3 (SWEC safety class). Class 3 is consistent with Group C Quality Standards required by Reg Guide 1.26 and SRP 3.2.2. The system also is designated Seismic Category I in accordance with the requirments of SRP 9.3.3, par. II.3.c; SRP 3.2.1; and Reg Guide 1.29.

The impact of the modifications of the aerated drain system within the safeguards building would be the addition of four safety-related, 12-20 gpm dewatering pumps, two 500 gpm drain pumps, and associated piping, valves, and instrumentation.

1.6 POST-ACCIDENT SAMPLING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the post-accident sampling system are:

<u>Standard Review Plans</u>

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

<u>Reg Guides</u>

Reg Guide 1.97, Rev.2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

<u>Other Related Documents</u>

NUREG 0737, November 1980, Clarification of TMI Action Plan Requirements

1.6.1 Detailed Discussion

Only one sampling location within the safeguards building would be part of the post-accident sampling system.

Reg Guide 1.97, Table 2, requires an installed capability to sample all ECCS pump sumps following an accident. This requirement applies to the four RS sump pumps in the safeguards building. The addition of sample tubing and valves to the discharge of the safeguards sump pumps would have a relatively small impact on the total post accident sample system and the safeguards building. Therefore, the system is addressed in its entirety in the discussion of the auxiliary building, where the major system components will be located.

1.7 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and effluent radiological monitoring and sampling systems are:

<u>Standard Review Plans</u>

SRP 9.3.2, Rev. 2, July 1981, Process Sampling System

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

1.7.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the safeguards building were reviewed to the above documents and found to be in basic compliance. System modifications described below would be required to achieve full compliance.

The safeguards ventilation exhaust no longer discharges to the common ventilation vent stack. SRP 11.5, Table 1A, Item 3, requires that continuous effluent radiation monitoring be provided for all individual building ventilation exhaust points. Therefore, gaseous and particulate radiation monitoring capability, with provisions for obtaining local grab samples, would be required for the safeguards building ventilation exhaust points. In addition, the safeguards building ventilation exhaust, as an identified release point, would be monitored for noble gas activity and vent flow rate over extended accident ranges as required by Reg Guide 1.97, Table 2. Also, sampling capability for particulates and halogens, with onsite analysis, would be required.

The impact of the above modifications to the process and effluent radiological monitoring and sampling systems is presented in the project position on Reg Guide 1.97.

1.8 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the radation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.8.1 Detailed Discussion

The radiation monitoring system in the safeguards building includes only area radiation monitoring. Post-accident, process, and effluent monitoring are covered in Sections A.1.7 and A.1.8.

The design of the area radiation monitoring system within the safeguards building has not been finalized. System modification will result in an impact.

As required by Reg Guide 1.97, Table 2, extended-range area radiation monitoring capability would be required for the safeguards building to aid in the assessment of the magnitude and consequences of postulated accidents and to provide data on the habitability of the building.

The impact of this system modification is presented in the position statement on Reg Guide 1.97.

A. SAFEGUARDS BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The detailed description of the system changes within the safeguards building is in Section 1. The general changes to the structure are described in this section.

The original safeguards building was a three-sided structure with no floors above the rock anchor gallery. None of its walls were adjacent to the containment.

The revised building is taller than the original building and has two floors on the top of the rock anchor gallery, an 18 in. thick wall adjacent to the containment, and a stairwell on the south side.

2.1 IMPACT DUE TO LICENSING CHANGES

Self-contained filters have been provided for the safeguards air filtration system in the safeguards building. Filtration of this area is required by SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System. Missile protection of the systems, including all ductwork, would be required by Reg Guide 1.117, Rev. 1, April 1978, Tornado Design Classification.

The radial walls in the pump cubicles have been increased in thickness from 12 in. to 24 in. to resist lateral forces due to internal flooding as required by SRP 3.4.1. A 12 in. thick concrete wall has been added at El 251 ft-9 in. to provide the separation required by Reg Guide 1.75, Rev.2, September 1978, Physical Independence of Electric Systems. A separate stair tower has been added to provide proper access and egress to the building.

2.2 3-D SEISMIC IMPACTS

The safeguards building would have to be redesigned for the 3-D seismic requirements of Reg Guides 1.60, 1.61, and 1.92, and SRPs 3.7.1 and 3.7.2. An 18 in. thick concrete wall has been added adjacent to the containment. The additional wall resists loads associated with 3-D seismic impacts by increasing the rigidity of the structure. The wall also acts as a barrier to contain groundwater and internal flood as required by SRP 3.4.1.

B. DECAY HEAT/QUENCH SPRAY BUILDING

SECTION 1

AREA SYSTEMS

1.1 DECAY HEAT REMOVAL SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the decay heat removal system (DHRS) are:

Standard Review Plans (SRP)

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 5.4.7, Rev. 2, July 1981, Residual Heat Removal (RHR) System

SRP 6.2.2, Rev. 3, July 1981, Containment Heat Removal Systems

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment)

1.1.1 Detailed Discussion

The portion of the DHRS located within the decay heat/quench spray (DH/QS) building was reviewed to the above documents and found to be in basic compliance, although several system modifications would be required to achieve full compliance.

SRP 6.3 establishes the design and testing procedures for the emergency core cooling systems (ECCS). The DHRS serves as part of the ECCS following a loss-of-coolant accident (LOCA) and must meet the requirements of SRP 6.3. That portion of the system which performs an ECCS function and is located in the DH/QS building will comply with SRP 6.3.

The DHRS is designated as a Class 2 system (SWEC safety class) which is consistent with Group B Quality Standards as required by SRP 5.4.7, SRP 3.2.2, and Reg Guide 1.26. The system is also consistent with Seismic Category I in accordance with the classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 5.4.7.

Reg Guide 1.139 and SRP 5.4.7 require the ability to safely take the reactor from an operating condition to cold shutdown by a method which meets single failure criteria and utilizes safety-related systems. Also, the system's safety function should be accomplished assuming the availability of only onsite or offsite power. In order to achieve these requirements, the normally locked closed valves outside the containment in the decay heat suction lines would be changed to motor-operated valves (MOV). This would permit remote operation capability from the control room. The modulating valves on the outlet of the decay heat (DH) coolers and bypass are air controlled. North Anna Unit 3 does not have a safety-related air supply; therefore, a new type modulating valve would be required.

In the DHRS, physical separation of redundant components and piping is required. The conceptual layout of the building would be revised to provide separation. The impact would be in relocating internal walls and components, and revising pipe routing. Two separate suction paths from the refueling water storage tank to the decay heat pumps would be required. Separation is required to accommodate a single failure during emergency cooling function.

The DHRS is designed to the 100 percent mechanial and electrical redundancy requirements for an engineered safety features (ESF) system as required by SRP 5.4.7. Confirmation that the postulated single failure (active or passive) would not result in a system loss-of-function would be documented in the system's Failure Modes and Effects Analyses Report. This report would be prepared per the requirements of SRP 5.4.7.

1.2 QUENCH SPRAY SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the quench spray (QS) system are:

<u>Standard Review Plans</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.2.2, Rev. 3, July 1981, Containment Heat Removal Systems

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Product Cleanup System

SRP 6.5.3, Rev. 2, July 1981, Fission Product Control Systems and Structures

<u>Reg Guides</u>

Reg Guide 1.1, Rev. 0, November 1970, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.2.1 Detailed Discussion

That portion of the QS system which exists in the DH/QS area was reviewed to the above documents and the system modifications described in this section would be required to achieve full compliance.

SRP 6.3 establishes the design and testing procedures for the ECCS. The QS system serves as part of the ECCS following a LOCA and must meet the requirements of SRP 6.3. That portion of the system which performs an ECCS function and is located in the safeguards building will comply with SRP 6.3.

The QS system is designated as a Class 2 system (SWEC safety class), which is consistent with Group B Quality Standards as required by SRP 6.2.2, SRP 3.2.2, and Reg Guide 1.26. The system is also consistent with Seismic Category I in accordance with the classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 6.2.2.

The requirements of SRP 6.2.2, par. II.7 relative to spray header and nozzle design are discussed elsewhere (refer to D.1.2).

A review of the QS piping for safety train separation identified several concerns. The 10 in. QS pump suction line for train B passes over the train A pump. A rerouting of the QS pump piping to provide safety train separation would be required.

The QS system is part of the containment heat removal system. It reduces the containment temperature by spraying 45 F water into the containment following a LOCA. The QS system is presently designed to the 100 percent mechanical and electrical redundancy requirements for an ESF. The system will be designed to accommodate an active single failure. Also, a failure modes and effects analysis of the system would be performed to ensure that the system is capable of withstanding all single active failures.

1.3 ESF COMPONENT COOLING WATER SYSTEM

Those documents reviewed which provide guidance and/or established criteria for the design basis of the ESF component cooling water (CCW) system are:

Standard Review Plan

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.2.2, Rev. 1, July 1981, Reactor Auxiliary Cooling Water Systems

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.3.1 Detailed Discussion

That portion of the CCW system which exists in the DH/QS building was reviewed to the above documents and found to be in basic compliance. System modifications described in this section would be required to achieve full compliance.

The ESF CCW system in the DH/QS building cools the reactor coolant water as it is circulated through the decay heat coolers. Two redundant supply and return headers would be required for separation of the safety flow path. The cross connect which previously interconnected both supply and return headers would be deleted to ensure no single or common mode failure could result in a loss of system function.

The CCW system was originally nonsafety-related and backed up the safety-related service water system. The CCW system was upgraded to be safety-related (refer to Section E.1.2) and the cross-connect with service water was deleted. The ESF component cooling, as it is safety-related and redundant, does not require the service water backup and provides a closed loop system to eliminate the potential release of reactor coolant in the event of a decay heat exchanger tube leak.

The ESF CCW supply and return headers piping is designated as a Class 3 system (SWEC safety class), which is consistent with Group C Quality Standards as required by SRP 9.2.2, par. II.3; Reg Guide 1.26; and, SRP 3.2.2. These portions of the system are also designated Seismic Category I in accordance with the classification requirements of SRP 9.2.1, par. II.8; SRP 3.2.1; and, Reg Guide 1.29.

1.4 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.5.1, Rev. 2, July 1981, Engineered Safety Feature Atmosphere Cleanup Systems

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

1.4.1 Detailed Discussion

The HVAC system for the decay heat removal area was reviewed to the above documents, and significant system modifications are required to achieve compliance. A discussion of these modifications is given below.

The previous design of the HVAC system in the decay heat removal building was part of the safeguards building ventilation system and consisted of two redundant supply fans and two redundant exhaust fans, with two common filter banks located in the auxiliary building. The system provided ESF atmosphere cooling by distributing outside air within the building and exhausting to the atmosphere.

The current HVAC system in the decay heat removal area consists of a secondary system used only during normal plant operation and a primary system used exclusively during transient plant conditions.

The secondary HVAC system consists of a nonsafety-related fan and associated ductwork and dampers. In those areas where this system may adversely impact safety-related systems, the equipment and ductwork will be seismically supported.

The HVAC system used during transient conditions consists of two redundant filter trains with associated fans that discharge to the atmosphere and provide ESF cleanup capability, a redundant source of chilled water to coolers, unit coolers which provide ESF equipment cooling capability, associated ductwork, dampers, and instrumentation.

The main reasons for the design changes outlined above are:

1. Reg Guide 1.52 Position C.2.f requires that the volumetric flow rate through a single cleanup train be limited to 30,000 cfm or below.

2. VEPCO has requested that maximum design outside air temperatures be increased to 107 FDB and 82 FWB.

With the higher temperature, the previous design (once-through ventilation) would require greater volumetric air flow in order to maintain an adequate ESF equipment ambient temperature. With a greater air flow, unacceptably large systems would result and multiple filters would be required to maintain flow through each filter train at below 30,000 cfm.

In addition, SRP 6.5.1 requires instrumentation for surveillance of the filter train operation.

1.5 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

<u>Standard Review Plans</u>

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

Reg Guide 1.52, Rev.2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.5.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In compliance with Reg Guide 1.52, a water spray system would be required for the charcoal filter assemblies.

A standpipe system with hose stations has been provided to protect the decay heat removal area.

1.6 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system (EFDS) are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

Reg Guides

Reg Guide 1.26, Rev. 1, September 1974, Quality Group Classifications and Standards for Water-, Steam-, and

Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.6.1 Detailed Discussion

The portion of the EFDS which exists in the DH/QS building was reviewed to the above documents. System modifications described in this section would be required to achieve full compliance.

The EFDS in the decay heat building consists of the decay heat area, floor drains piping network, and the aerated drain system. As the decay heat building will be redesigned to allow separation of redundant components, the aerated drain system will also require some modification in order to comply fully with the SRP 9.3.3 criteria. The floor drains would require modification from the design shown on the current building service drawings.

An additional sump and pumps would be required to maintain the necessary decay heat area separation. One sump and two sump pumps have been dedicated to this area. Each of the existing pumps from Units 3 and 4 would be used.

1.7 POST-ACCIDENT SAMPLING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the post-accident sampling system are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

Reg Guides

Reg Guide 1.97, Rev.2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.7.1 Detailed Discussion

Reg Guide 1.97, Table 2, requires an installed capability to sample all ECCS pump sumps following an accident. This requirement applies to the decay heat removal sump pumps in the DH/QS building. The impact associated with adding sample tubing and valves to the discharge of the decay heat removal sump pumps would have a relatively small impact on the total redesign of the post accident sample system and the DH/QS building. Therefore, the system and its impact will be addressed in its entirety in Section E, Auxiliary Building, where the major system components will be located.

1.8 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and post-accident sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guides

Reg Guide 1.97, Rev. 3, July 1981, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.8.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the DH/QS building were reviewed to the above documents and found to be in basic compliance. System modifications described below would be required to achieve full compliance.

As documented in Section I.1, the DH/QS ventilation exhaust no longer discharges to the common ventilation vent stack. SRP 11.5, Table 1A, Item 3, requires that continuous effluent radiation monitoring be provided for all individual building ventilation exhaust points. Therefore, gaseous and particulate radiation monitoring capability, with provisions for obtaining local grab samples, would be required for the DH/QS building ventilation exhaust points.

In addition, the DH/QS building ventilation exhaust, as an identified release point, would be monitored for noble gas activity and vent flow rate over extended accident ranges, as required by Reg Guide 1.97, Table 2. Also, sampling capability for particulates and halogens, with onsite analysis, would be required.

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment), requires that cooling water radioactivity be monitored at the output of the decay heat removal (DHR) heat exchangers. Component cooling water is monitored for radioactivity.

In addition, to satisfy the Reg Guide 1.139 requirement, two offline liquid radiation monitors would be required to monitor the component cooling water flow from each decay heat removal heat exchanger. These monitors will be located at El 255 ft of the DH/QS structure over the rock anchor gallery.

1.9 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the area radiation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.9.1 Detailed Discussion

The radiation monitoring system in the DH/QS building includes only area radiation monitoring. Post-accident, process, and effluent monitoring are covered in Sections A.1.7 and A.1.8.

The design of the area radiation monitoring system within the DH/QS building has not been finalized.

Presently, as required by Reg Guide 1.97, Table 2, extended-range area radiation monitoring capability would be required for the DH/QS building to aid in the assessment of the magnitude and consequences of postulated accidents and to provide data on the habitability of the building.

Because of the relatively low post-accident ambient radiation dose levels and close proximity to other process and effluent streams which require normal and post-accident radiation monitoring, the following monitors would also be located over the rock anchor gallery in the DH/QS building:

- Four recirculation spray heat exchanger service water outlet radiation monitors

- Main steam offline radiation monitors

- DH/QS ventilation exhaust radiation monitors
- Safeguards ventilation exhaust radiation monitors

B. DECAY HEAT/QUENCH SPRAY BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The main steam valve house (MSVH) and decay heat/quench spray (DH/QS) area form a single structure. The description of the structural changes for these areas are treated as a single unit.

2.1 IMPACT DUE TO LICENSING CHANGES

2.1.1 Main Steam Valve House

The details of the system changes within the building are presented in Section C.1.

Interior walls have been added and existing walls have been relocated. This has necessitated reframing all floors. Interior pipe chase and stairwells have been added. An east and west pipe tunnel has been added below the building. Vent openings that were originally located in the east and west walls have been relocated to the north wall of the MSVH.

The major impact to the building below El 284 is due to Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems, and to fluid system changes required to address passive failure and post-accident access, i.e., shielding. The impact to the building above El 284 is relocation of vent areas from the east and west side of the building to the north side.

2.1.2 Decay Heat/Quench Spray Building

The details of the system changes associated with the DH/QS building are in Section B.1.

Interior walls have been added and existing walls have been relocated. This has necessitated reframing all floors. Interior pipe chases and stairwells have been added. An east and west pipe tunnel has been added below the building. Two additional floor elevations and a rock anchor gallery with stairwells to the east and west have been added to the DH/QS area. A concrete slab has replaced a steel framed floor elevation in the DH/QS area.

The major impact to the building below El 284 is due to Reg Guide 1.75 as well as fluid system changes required to address passive failure and post-accident access i.e., shielding.

Additional requirements of more space, increased height, and stairwells at the east and west ends of the building are a result of providing a self-contained air filtration system in the DH/QS area, as required by SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System.

2.2 3-D SEISMIC IMPACTS

The MSVH and DH/QS building would have to be redesigned for 3-D seismic requirements per Reg Guides 1.60, 1.61, and 1.92, and SRPs 3.7.1 and 3.7.2.

C. MAIN STEAM VALVE HOUSE

SECTION 1

AREA SYSTEMS

1.1 MAIN STEAM SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the main steam (MS) supply system are:

Standard Review Plans (SRP)

SRP 6.2.4, Rev. 2, July 1981, Containment Isolation System

SRP 10.3, Rev. 2, July 1981, Main Steam Supply System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Other Related Documents

BTP MTEB 5-3, Rev. 2, July 1981, Monitoring of Secondary Side Water Chemistry in PWR Steam Generators

1.1.1 Detailed Discussion

Although portions of the MS system are located in various areas of the facility, all licensing impacts associated with the MS system are included in this section.

The MS system consists of all components from the steam generator nozzles up to and including the turbine stop valve and the steam supply system for the turbine-driven auxiliary feedwater pump.

The MS system was reviewed to the above licensing documents and found to be in basic compliance, although several system modifications would be required to achieve full compliance.

The air-operated modulating atmospheric dump (MAD) valves, supplied by B&W and located in the main steam valve house (MSVH), should be modified to operate in the long term such that cold shutdown can be achieved and maintained using only safety-grade components and power sources, as required by SRP 10.3, par. III.5.f. Currently, the compressed air system is a nonsafety-related system.

The addition of two MAD valves to the MS system, for a total of four MAD valves, would be required to comply with the single failure requirements of SRP 10.3, par. III.5.d. This will result in a piping revision associated with the existing MAD valves.

As required by SRP 10.3, par. III.6, hazards evaluations and failure modes and effects analyses of the MS system would be required to verify that:

- Failure of non-Seismic Category I portions of the MS system, or of other systems located close to essential portions of the system, do not preclude operation of the essential portions of the MS system.

- Essential functions of the system can be maintained in the event of adverse environmental phenomena, certain pipe ruptures, and loss of offsite power.

The safety-related portions of the MS system are designated Class 2 (SWEC safety class), which is consistent with Group B Quality Standards as required by SRP 10.3, par. III.3; SRP 3.2.2, Appendix A; and Reg Guide 1.26. These and other portions of the MS system are designated Seismic Category I in accordance with the classification requirements of Reg Guide 1.29 and SRP 10.3, par. III.3.

1.2 FEEDWATER SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design of the feedwater system are:

<u>Standard Review Plans</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 10.4.7, Rev. 2, July 1981, Condensate and Feedwater System

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards of Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.2.1 Detailed Discussion

Although portions of the feedwater system are located in various areas of the facility, all licensing impacts associated with the feedwater system are included in this section.

Based on a review of the feedwater system to the above licensing documents, the feedwater system is in basic compliance, although several analyses would be required to document full compliance.

Hazards evaluations and failure modes and effects analyses of the feedwater system would be required to verify that essential functions of the system can be maintained in the event of adverse environmental phenomena, certain pipe ruptures, and loss of offsite power, as required by SRP 10.4.7, par. III.1.

The safety-related portions of the feedwater system are designated Class 2 (SWEC safety class), which is consistent with Group B Quality Standards as required by SRP 10.4.7, par. III.3, and Reg Guide 1.26. These portions of the system are also consistent with Seismic Category I in accordance with the classification requirements of Reg Guide 1.29 and SRP 10.4.7, par. II.1.

1.3 AUXILIARY FEEDWATER SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the auxiliary feedwater system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

SRP 10.4.9, Rev. 2, July 1981, Auxiliary Feedwater System (PWR)

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group

Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Other Related Documents

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

1.3.1 Detailed Discussion

Although portions of the auxiliary feedwater system are located in various areas of the facility, all licensing impacts associated with the auxiliary feedwater system are included in this section.

The auxiliary feedwater system was reviewed to the above licensing documents and several system modifications would be required to achieve full compliance.

The auxiliary feedwater system is designed to the 100 percent electrical and mechanical redundancy requirements for an engineered safety features system, as required by SRP 10.4.9, par. II.5.b. This is achieved by providing adequate electrical separation as required by Reg Guide 1.75 and fire protection as required by SRP 9.5.1 and BTP CMEB 9.5-1.

As required by SRP 10.4.9, par. III.2, III.3, and III.4, hazards evaluations, failure modes and effects analyses, and reliability analyses of the auxiliary feedwater system would be required to verify that essential functions of the system can be maintained in the event of adverse environmental phenomena, certain pipe ruptures, and loss of offsite power.

The auxiliary feedwater system is designated Class 2 and Class 3 (SWEC safety classes), which is consistent with Group B and Group C Quality Standards as required by SRP 10.4.9, par. II.1 and Reg Guide 1.26. The system is also designated Seismic Category I in accordance with the classification requirements of SRP 10.4.9, par. III.1.C and Reg Guide 1.29.

Also, to enhance physical separation of the auxiliary feedwater system components, a new structure should be provided adjacent to the reactor containment. The new structure would house the motor-driven auxiliary feed pumps and associated valves and piping. The existing area in the MSVH would be dedicated to the steam driven auxiliary feedwater pumps.

1.4 STEAM GENERATOR BLOWDOWN SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the steam generator blowdown system (SGBS) are:

<u>Standard Review Plans</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 10.4.8, Rev. 2, July 1981, Steam Generator Blowdown System (PWR)

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.143, Rev. 1, October 1979, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

1.4.1 Detailed Discussion

Although portions of the SGBS are located in various areas of the facility, all licensing impacts associated with the SGBS are included in this section.

The SGBS was reviewed to the above licensing documents and found to be in basic compliance, although several system modifications would be required to achieve full compliance.

Currently, a piping run of the SGBS penetrates the reactor containment into the auxiliary building and then is routed through the east wall into the MSVH. This segment of the SGBS provides secondary sampling connections and is a high energy line. In order to minimize hazards to safety systems in the auxiliary building, this piping run should be relocated so that it penetrates the reactor containment directly into the MSVH.

The steam generator blowdown system is designated Class 2 and Class 4 (SWEC safety classes), which is consistent with Group B and Group D Quality Standards as required by SRP 10.4.8, par. II.4, Reg Guide 1.143, par. C.1.1, and Reg Guide 1.26. Portions of the SGBS designated Class 2 are also designated

Seismic Category I in accordance with the classification requirements of Reg Guide 1.29 and SRP 10.4.8, par. III.1.d.

1.5 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.5.1 Detailed Discussion

The HVAC system for the MSVH was reviewed to the above documents and tornado dampers in the auxiliary feedwater pump area inlet and outlet openings should be provided. In addition, modifications to the HVAC system are required due to an increase in maximum outside temperature to 107 FDB and 82 FWB.

The previous design of the HVAC system in the auxiliary feedwater pump area of the MSVH consisted of two redundant supply fans and two redundant exhaust fans. The system provided ESF equipment atmosphere cooling by distributing outside air through the building and exhausting to the atmosphere. The remaining area within the MSVH depended on natural (gravity) ventilation for cooling.

The current HVAC system in the auxiliary feedwater pump area of the MSVH consists of a secondary system used only during normal plant operation and a primary system used exclusively during transient plant conditions.

The secondary HVAC systems consists of nonsafety-related fans, and associated ductwork and dampers. In those areas where this system may adversely impact safety-related systems, the equipment and ductwork will be seismically supported.

The HVAC system used during transient conditions consists of unit coolers which provide engineered safety feature (ESF) equipment cooling capability for areas housing ESF equipment within the auxiliary feedwater pump area, and associated ductwork, dampers, and instrumentation.

1.6 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979

1.6.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of SRP 9.5.1 and BTP CMEB 9.5-1. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The MAD valves should be separated to comply with the criteria advanced in BTP CMEB 9.5-1.

1.7 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the radation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.7.1 Detailed Discussion

The area radiation monitoring system in the MSVH includes only area radiation monitoring. Process and effluent monitoring is covered in Section C.1.8. Post-accident sampling requirements do not apply to the MSVH.

The design of the area radiation monitoring system within the MSVH has not been finalized. The system will comply with the above documents.

As required by Reg Guide 1.97, Table 2, extended-range area radiation monitoring capability would be required for the MSVH to aid in the assessment of the magnitude and consequences of postulated accidents and to provide data on the habitability of the building.

1.8 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and effluent radiological monitoring and sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process Sampling System

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.8.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the MSVH were reviewed to the above documents and found to be in basic compliance. System modifications described below would be required to achieve full compliance.

As required by Reg Guide 1.97, Table 2, Type E Variables, effluent radiation monitoring for noble gases and mass flow rate indication would be required for the exhaust from the steam generator safety relief valves and the MAD valves.

In addition, the MS exhausted from the turbine driven auxiliary feedwater pump, as an identified release point, would also require the necessary instrumentation for monitoring noble gas activity and mass flow rate in accordance with the requirements of Reg Guide 1.97, Table 2.

The impact of the above modifications is contained in the project position statement on Reg Guide 1.97.

1.9 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the equipment and floor drainage system are:

<u>Standard Review Plan</u>

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

<u>Reg Guide</u>

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.9.1 Detailed Discussion

The portions of the equipment and floor drainage system which exist in the MSVH were reviewed to the above documents and the following modifications would be required to achieve full compliance.

Where required by SRP 9.3.3, Class 3 piping would be used instead of Class 4. An associated engineering and design effort is required to make necessary changes in current pipe and flow diagrams and support documents.

D. REACTOR CONTAINMENT

SECTION 1

AREA SYSTEMS

1.1 CONTAINMENT HEAT REMOVAL SYSTEM (RECIRCULATION SPRAY SYSTEM)

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the recirculation spray (RS) system are:

<u>Standard Review Plans (SRP)</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.2.1, Rev. 2, July 1981, Containment Functional Design

SRP 6.2.1.1A, Rev. 2, July 1981, PWR Dry Containments, including Subatmospheric Containments

SRP 6.2.1.3, Rev. 1, July 1981, Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

SRP 6.2.1.4, Rev. 1, July 1981, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

SRP 6.2.1.5, Rev. 2, July 1981, Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

SRP 6.2.2, Rev. 3, July 1981, Containment Heat Removal Systems

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Cleanup System

<u>Reg Guides</u>

Reg Guide 1.1, Rev. 0, November 1970, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.82, Rev. 0, June 1974, Sumps for Emergency Core Cooling and Containment Spray Systems

1.1.1 Detailed Discussion

That portion of the RS system which exists in the reactor containment consists of the reactor containment sump and spray headers. These portions of the system were reviewed and the system modifications described in this section would be required to achieve full compliance.

Redundant, 100 percent capacity, RS headers are inside the reactor containment. To produce spray patterns which maximize the containment volume covered and minimize the overlapping of sprays as required by SRP 6.2.2, par. II.3; the number, type, and orientation of the spray nozzles are being revised.

The following modifications would be required to the design of the reactor containment sump, based on Reg Guide 1.82.

The four screened enclosures presently around each sump will be retained. The sump pit should be separated into two screened enclosures, one for each pair of redundant pump suctions. Each enclosure should be completely separate from the other and should be equipped with its own trash rack. Each trash rack should not only extend in front of each enclosure, but should extend in front of each of the adjacent enclosures. This will provide two sets of trash racks and screens between the redundant halves of the sump. In addition, the sump design should be modified as required to protect the sump from damage to the intake filters by whipping pipes and high velocity jets of water and steam.

The screens should be raised to a level at or above floor elevation. Both the outer coarse screen and inner fine screen will be retained for each enclosure.

Suction pipe openings should be raised and the slope of the floor altered to slope away from the suction pipe openings toward the center of the containment.

The maximum coolant velocity value of 0.2 ft/sec is exceeded until the screens are fully submerged. Therefore, the redesign effort will strive to minimize this velocity within the physical constraints of the existing containment arrangement. To further reduce the possibility of entraining debris into the suction piping, a trench 18 in. wide and 18 in. deep should be provided around the outside of the trash racks of both enclosures. This trap would reduce the amount of debris which impinges on the screens.

No major rework of the general arrangement of the containment is contemplated.

1.2 QUENCH SPRAY SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the quench spray (QS) system within the reactor containment are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Product Cleanup System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.2.1 Detailed Discussion

The portion of the QS system located within the reactor containment was reviewed to the above documents and found to be in basic compliance, although some system modifications would be required to achieve full compliance.

The portion of the QS system in the reactor containment consists of the QS headers. In order to meet the requirements of SRP 6.5.2, par. II.1.b, and assure full coverage of the containment volume, two QS headers would be required. One header would be located high in the containment dome and the other would be located in the lower section of the dome. The number, type, and orientation are to be revised.

1.3 REACTOR COOLANT SYSTEM

Those documents reviewed which provide and/or establish criteria for the design basis of the reactor coolant system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 5.4.11, Rev. 2, July 1981, Pressurizer Relief Tank

SRP 5.4.12, Rev. 0, July 1981, Reactor Coolant System High Point Vents

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification Reg Guide 1.45, Rev. 0, May 1973, Reactor Coolant Pressure Boundary Leakage Detection System

Reg Guide 1.45, Rev. 0, May 1973, Reactor Coolant Pressure Boundary Leakage Detection System

Reg Guide 1.147, Rev. 0, February 1981, Inservice Inspection Code Case Acceptability ASME Section XI, Division 1

1.3.1 Detailed Discussion

The reactor coolant system was reviewed to the above documents. The following modifications are recommended.

SRP 5.4.12 requires high point vents to exhaust non-condensible gases from the primary coolant system that could inhibit natural circulation flow. Additional piping and solenoid-operated valves would be required. Non-condensible gases would exhaust to the reactor coolant drain tank or to the containment.

SRP 5.4.11 requires the reactor coolant drain tank system (RCDTS) be designed so that the system function can be maintained as required in the event of a loss of offsite power. The RCDTS is designed as a nonseismic, nonsafety-related, QA Category II system. Therefore, in the present design, the RCDTS will not operate in the event of a loss of offsite power. The RCDTS should be designed to permit operability of the system in the event of a loss of offsite power.

Reg Guide 1.45 requires that the reactor coolant leakage detection system be capable of performing its function following seismic events not requiring plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to a safe shutdown earthquake. Additional review and analysis is required. The reactor coolant leakage detection system should be equipped to permit calibration and testing for operability.

1.4 DECAY HEAT SYSTEM

Those documents reviewed which provide and/or establish criteria for the design basis of the decay heat system within the reactor containment are:

Standard Review Plan

SRP 5.4.7, Rev. 2, Residual Heat Removal (RHR) System

Reg Guide

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment)

1.4.1 Detailed Discussion

The portion of the decay heat system which exists inside the reactor containment consists of the two suction lines and associated reactor coolant isolation valves. In the decay heat removal system (DHRS), the interlocks should be independent and diverse to prevent the opening of the DHRS/reactor coolant system (RCS) suction isolation valves if the RCS pressure is above the DHRS design pressure. The valves should also receive a signal to close automatically whenever the RCS pressure exceeds the DHRS design pressure.

Reg Guide 1.139 and SRP 5.4.7 require two or more independent interlocks for the suction isolation valve system to automatically close or prevent opening should the RCS pressure exceed the DHRS design pressure. The North Anna Unit 3 DHRS isolation valve system is in partial compliance with the above requirements. Full compliance would be achieved by supplying independent interlock and control power to each series suction isolation valve in each parallel path. This would be accomplished by replacing two of the four 480 V motor operators with 120 V operators powered from independent 120 V vital buses. In order to provide necessary separation of the redundant suction lines and associated electrical cables, these lines would be rerouted and isolation valves relocated.

1.5 ENGINEERED SAFETY FEATURE COMPONENT COOLING WATER SYSTEM

Those documents reviewed which provide and/or establish criteria for the design basis of the engineered safety feature (ESF) component cooling water system in the reactor containment are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 2, July 1981, System Quality Group Classification

SRP 9.2.2, Rev. 1, July 1981, Reactor Auxiliary Cooling Water Systems

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment)

1.5.1 Detailed Discussion

The portion of the ESF component cooling water system within the reactor containment consists of the cooling water piping serving the reactor coolant pumps and motors and the letdown coolers.

In the reactor containment, the original component cooling was nonsafety-related. The component cooling would be upgraded to be safety-related (refer to Section E.1.2) and would have redundant supply and return headers. The new piping arrangement would require deleting the existing six, 6 in. diameter containment penetrations and adding four, 8 in. diameter containment penetrations. The impact will be the four new penetrations and piping redesign within the reactor containment.

1.6 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC systems are:

<u>Standard Review Plan</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

<u>Reg Guides</u>

Reg Guide 1.140, Rev. 1, October 1979, Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.6.1 Detailed Discussion

The HVAC systems for the reactor containment were reviewed to the above documents and equipment modifications would be required to achieve compliance. A discussion of these modifications is given below.

In the previous design, each of the two charcoal filter assemblies for the iodine filtration system located inside the reactor containment consisted of prefilters and a bank of high efficiency particulate filters upstream and downstream of the activated charcoal adsorbers.

Two filter assemblies would be required in the current design of the iodine filtration system to meet the Reg Guide 1.140 requirements as to the number, arrangement, and location of filter components, and filter construction, access, and clearances. Each assembly would be equipped with prefilters, HEPA filters upstream and downstream of the charcoal adsorbers, and heating coils to control the humidity before filtration.

In the previous design, the reactor containment purge exhaust was provided by the common exhaust system, which was equipped with two 64,000 cfm charcoal filter assemblies. Since Reg Guide 1.140 limits the filter assembly capacity to 30,000 cfm, the common exhaust system has been replaced by individual exhaust systems for each of the buildings which were connected to the common exhaust system.

In the current design, the containment purge exhaust system would be required with two 18,000 cfm capacity charcoal filter assemblies. Each assembly would consist of a centrifugal fan, demister, heating coil, and a bank of high efficiency particulate filters upstream and downstream of the activated charcoal adsorbers.

1.7 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

<u>Standard Review Plans</u>

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

<u>Reg Guides</u>

Reg Guide 1.141, Rev. 1 (draft 2), October 1979, Containment Isolation Provisions for Fluid Systems

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979

1.7.1 Detailed Discussion

The review of the fire protection system was completed to the requirements stated in SRP 9.5.1 and BTP CMEB 9.5-1. Reg Guide 1.120, Rev. 1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design the reactor containment fire protection system consisted of a standpipe system with hose racks and water spray systems for the charcoal filter assemblies.

In compliance with the guidelines of BTP CMEB 9.5-1, an oil leak collection system for the reactor coolant pumps would be required to the present fire protection system design. This system would be designed and installed to withstand a safe shutdown earthquake.

An automatic suppression system for the cable penetration area in the reactor containment has been identified as a licensing impact. This system was advanced based on the provisions for fire protection presented in Appendix R. BTP CMEB 9.5-1 has deleted the reference to automatic suppression as an acceptable method of fire protection within the containment. It introduces the concept of a 1/2 hour radiant energy shield. It represents the only method of reactor containment fire protection compatible with the present design. SWEC is in the progress of assessing the significance of this change. A recommendation will be provided separately.

1.8 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

Reg Guides

Reg Guide 1.26, Rev. 1, September 1974, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 1, August 1973, Seismic Design Classification

1.8.1 Detailed Discussion

The portion of the equipment and floor drainage system (EFDS), which exists in the reactor containment was reviewed to the above documents and found to be in basic compliance. Only minor systems modifications would be required to achieve full compliance.

1.9 POST-ACCIDENT SAMPLING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the post-accident sampling system are:

Standard Review Plan

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.9.1 Detailed Discussion

Three sampling locations within the reactor containment building will be part of the post-accident sampling system.

Reg Guide 1.97, Table 2, and SRP 9.3.2 require an installed capability to sample RCS, containment sump, and containment atmosphere following an accident. System modifications will be required to obtain each of these three post-accident samples. These samples will require a modification to include the necessary branch sample tubing and remotely controlled valves. These modifications will be a relatively small fraction of the total post-accident sampling system. Accordingly, the system will be addressed in its entirety in the review of the auxiliary building, where the major system components are located.

1.10 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and effluent monitoring and sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.10.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the reactor containment were reviewed to the above documents and found to be in full compliance.

1.11 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the radiation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.11.1 Detailed Discussion

The radiation monitoring system in the containment building includes:

- Airborne gas and particulate radiation monitoring
- Area radiation monitoring
- Containment purge exhaust radiation monitoring

Post accident, process, and effluent monitoring are covered in Sections D.1.9 and D.1.10.

The design of the area radiation monitoring system within the containment building has not been finalized. As required by Reg Guide 1.97, Table 2, extended-range area radiation monitoring capability would be required for the containment building to aid in the assessment of the magnitude and consequences of postulated accidents.

D. REACTOR CONTAINMENT

SECTION 2

BUILDING IMPACT ASSESSMENT

2.1 REACTOR CONTAINMENT INTERNAL STRUCTURE

2.1.1 Impact Assessment

The details of building changes are in Section D.1. The general changes to the structure are as follows.

2.1.1.1 Building Impact Assessment of Containment Internals

There are no major structural changes to the internal structures. Embedments of some of the major equipment supports will have to be redesigned due to revised Babcock & Wilcox (B&W) loads. The LOCA and seismic analysis of the structure would have to be reanalyzed and redesigned.

2.1.2 Impact Due to Licensing Changes

All reanalysis and redesign mentioned in Section 3.1.1 is required due to licensing changes caused by the following:

- Commitment to 3-D seismic impacts as defined by Reg Guides 1.60, 1.61, and 1.92

- Commitment to perform asymmetric cubicle pressurization analyses per NUREG 0609

- Revised B&W loads per NUREG 0609

2.1.3 3-D Seismic Impacts

The 3-D seismic requirements involve a check of the adequacy of the design of the internals structure. No concrete outline changes are anticipated.

2.2 REACTOR CONTAINMENT LINER

2.2.1 Impact Assessment

The RC liner requires several modifications due to licensing changes and 3-D seismic requirements. These modifications are primarily the result of changes to the liner penetrations. It is estimated that approximately 94 of the 128 required penetrations would either require a new penetration or some rework. Thirty-two new electrical penetrations have been added to the liner. Liner inserts in the location of the new electrical penetrations would have to be revised to correct interferences. Also, the dome liner would have to be revised to satisfy new spray system licensing requirements which would result in one additional circular spray header.

2.2.2 Impact Due to Licensing Changes

Licensing changes have caused major revisions to the RC liner piping penetrations. There were previously 66 penetrations in the penetration banks entering the auxiliary building. Now 72 are required. It is estimated that all 72 would require either a new penetration or some degree of rework. In the main steam valve house area there are currently 15 penetrations. After the changes are incorporated, there would be 27 penetrations, of which approximately 15 would require either a new penetration or rework. In the decay heat/quench spray (DH/QS) area, eight of the nine existing penetrations would require new penetrations or rework. No additional penetrations are added to this area. The safeguards area required six additional penetrations. The eight existing penetrations would remain unchanged. There are currently 70 electrical penetrations in the auxiliary building; thirty-two electrical penetrations would be added due to licensing changes.

Some liner inserts would also require revisions where interferences have been created. Most of these changes are based on one or more of the following documents:

SRPs 9.2.3, 9.3.3, and 9.3.4	Separation
SRP 3.4.1, Reg Guide 1.102	Flooding
SRP 3.5.1.1, Reg Guide 1.115	Internal Missiles
SRPs 3.6.1 and 3.6.2	Pipe Break
SRP 9.5.1	Fire Protection
NUREG 0737	Post Accident Sample System
NUREG 0800	Concrete Containment (SRP 3.8.1)
Reg Guides 1.60, 1.61, and 1.92	3-D Seismic Requirements

The requirement for two quench spray headers presented in SRP 6.5.2, Rev 1, July 1981, Containment Spray as a Fission Product Cleanup System, would result in revisions to the dome liner analysis and design.

2.2.3 3-D Seismic Impacts

A wall has been added adjacent to the RC in the DH/QS building to satisfy 3-D seismic requirements. The addition of this wall makes it necessary to install new extended penetrations in this area. Analysis will have to be performed on these new, sleeved penetrations. All penetrations should be checked to verify acceptability due to the 3-D seismic loading condition.

2.3 REACTOR CONTAINMENT EXTERIOR

2.3.1 Impact Assessment

The RC shell will require some modifications due to penetration changes. The RC shell, mat, and ring girder should be reanalyzed for the 3-D seismic loading condition. Liner penetration revisions required by licensing and 3-D changes will cause local rearrangement and analysis of reinforcing steel in the exterior shell.

2.3.2 Impact Due to Licensing Changes

In the exterior shell of the containment, some new or relocated penetrations will cause local rearrangement of reinforcing steel (rebar) where interferences have been created. The ring girder design requires verification for 3-D seismic loading. This will involve drawing changes and reanalysis of the rebar around these penetrations.

E. AUXILIARY BUILDING

SECTION 1

AREA SYSTEMS

1.1 MAKEUP AND PURIFICATION SYSTEM, INCLUDING CHEMICAL ADDITION

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the makeup and purification system are:

<u>Standard Review Plans (SRPs)</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

SRP 9.3.4, Rev. 2, July 1981, Chemical and Volume Control System (PWR) (including boron recovery system)

<u>Reg Guides</u>

Reg Guide 1.1, Rev. 1, December 1970, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment)

1.1.1 Detailed Discussion

The portion of the makeup and purification system located within the auxiliary building was reviewed to the above documents and several system modifications would be required to achieve full compliance.

The makeup and purification system is designed to the 100 percent mechanical and electrical redundancy requirements for an engineered safety features (ESF) system as required by SRP 9.3.4, par. II. Confirmation that the postulated single failure (active or passive) will not result in a system loss-of-function would be documented in the system's Failure Modes and Effects Analyses Report. This report would be prepared per the requirements of SRP 9.3.4, par. III.

Reg Guide 1.139 requires the ability to take the reactor safely from an operating condition to cold shutdown by a method which meets single failure criteria and utilizes safety-related systems. The system safety function should be accomplished assuming the availability of only onsite or offsite power. In order to provide pressurizer control during loss of power a safety-related source of high pressure spray would be required. A line from each of the redundant makeup high pressure injection lines would be connected to the normal pressurizer spray line. This would provide safety-related high pressure spray from redundant safety trains.

In order to provide safety-related boration of the reactor coolant system, the chemical addition system should be upgraded to be safety-related. Two ASME III C.2, Quality Group B, Seismic Class I, boric acid tanks and pumps would be required. Each boric acid tank and pump would be connected to the suction of the makeup pumps of its associated safety-related train.

The redundant safety-related portion of the makeup system has been rearranged to provide physical and electrical separation. The safety trains are cross-connected and isolated by two redundant isolation valves. This involves using one of Unit 4's makeup pumps so each safety train has a spare pump to allow maintenance and testing during power operation and to satisfy separation requirements.

Differential pressure gauges would be required across the purification and deborating demineralizers inlet and discharge lines as required by SRP 9.3.4. These gauges would provide an indication of resin bed condition in addition to the planned effluent analyses and radiation surveys.

As required by SRP 9.3.4 and Reg Guide 1.1, the makeup pumps have been designed so as to ensure adequate net positive suction head under all modes of makeup system operation.

SRP 6.3 establishes the design and testing procedures for the emergency core cooling systems (ECCS). The makeup and purification system serves as part of the ECCS during the long term following a loss-of-coolant accident (LOCA) and should meet the requirements of SRP 6.3. That portion of the system which performs an ECCS function and is located in the auxiliary building will comply with SRP 6.3.

The makeup and purification spray system is designated as a Class 2 system (SWEC safety class), which is consistent with Group B Quality Standards as required by SRP 9.3.4, par. II, SRP 3.2.2, and Reg Guide 1.26. The system is also consistent with the Seismic Category I classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 9.3.4, par. II.

1.2 ESF COMPONENT COOLING WATER SYSTEMS

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the component cooling water (CCW) systems are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.2.2, Rev. 1, July 1981, Reactor Auxiliary Cooling Water Systems

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electrical Systems

Reg Guide 1.139, Rev. 0, May 1978, Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown (for comment)

1.2.1 Detailed Discussion

The portions of the component cooling water systems located within the auxiliary building were reviewed to the above documents and several system modifications would be required to achieve full compliance.

SRP 9.2.2 provides criteria for the review of cooling systems for safety-related components. The ESF CCW system removes process heat from components and transfers this heat to the service water system. Detection of radioactivity within the system, separation of safety-related components from the nonsafety-related components, and redundancy between safety-related components have been provided. The CCW system, which was originally designed to remove heat from components of two reactor units, will now service only North Anna Unit 3.

This ESF CCW system was not designed to be a safety-related system and was not completely redundant and separate or single failure-proof. When Reg Guide 1.139 was reviewed, it was determined that the CCW system should be upgraded to meet its requirements for safe cold shutdown. To achieve this, the CCW system would have to be divided into two redundant safety-related trains and a single nonsafety-related train, with no cross-connections between the three trains. This complete separation and redundancy would ensure that safety functions would be provided during normal and accident conditions, thereby meeting the requirements of SRP 9.2.2 and Reg Guide 1.139.

The major components and piping of the CCW systems are located in the auxiliary building. The pumps, surge tanks, and heat exchangers are to be rearranged in the auxiliary building to provide separation of the redundant trains in the ESF CCW system. Also, the CCW system (nonsafety) would be rearranged so that it is separated from the ESF CCW redundant trains. The major components added to the systems would be two nonsafety CCW pumps, one ASME III surge tank, and one nonsafety ASME VIII surge tank. All existing equipment from the Units 3 and 4 CCW system, except for one heat exchanger, would be used.

The CCW system (nonsafety) will be designed to Class 4 (SWEC nonnuclear classification), ANSI B31.1, ASME VIII, and SRP 9.2.2 requirements.

The ESF CCW system is designed to the redundant 100 percent mechanical and electrical requirements of SRP 6.2.2, par. II. Confirmation that the postulated single failure (active or passive) will not result in a system loss-of-function would be documented in the system's Failure Modes and Effects Analyses Report. This report would be prepared per the requirements of SRP 9.2.2, par. III.

The ESF CCW system is designated as a Class 3 system (SWEC safety class), which is consistent with Group C Quality Standards as required by SRP 9.2.2, par. II; SRP 3.2.2; and Reg Guide 1.26. The system is also consistent with Seismic Category I in accordance with the classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 9.2.7, par. II.

1.3 POST-LOCA HYDROGEN RECOMBINER SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the post-LOCA hydrogen recombiner system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.2.5, Rev. 2, July 1981, Combustible Gas Control in Containment

Reg Guides

Reg Guide 1.7, Rev. 2, November 1978, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Other Related Documents

NUREG 0737, November 1980, Clarification of TMI Action Plan Requirements

1.3.1 Detailed Discussion

That portion of the post-LOCA hydrogen recombiner system which exists in the auxiliary building was reviewed to the documents identified above. System modifications described in this section would be required to achieve full compliance.

The post-LOCA hydrogen recombiner system is designed to process containment atmosphere through redundant hydrogen recombiner trains. Hydrogen and oxygen are thermally combined to form water, thereby reducing the hydrogen concentration and preventing gases from combusting. Dedicated hydrogen recombiners would be obtained and would be located permanently within the auxiliary building. They would take suction from and discharge to the containment as required by NUREG 0737, II.E.4.2.

The post-LOCA hydrogen recombiner system is designed to the 100 percent mechanical and electrical requirements for an ESF system as required by SRP 6.2.5, par. II. Confirmation that the postulated single failure (active or passive) would not result in a system loss-of-function would be documented in the system's Failure Modes and Effects Analyses Report. This report would be prepared per the requirements of SRP 6.2.5, par. II.

The hydrogen recombiner would be rearranged to provide adequate separation of the redundant trains and separation from nonsafety-related equipment in the auxiliary building.

SRP 6.2.5 (reference NUREGs 0737 and 0718) recommends that plants utilizing external recombiners have dedicated containment penetrations. Two reactor containment penetrations would be required with sufficient provisions to assure containment isolation when required.

The post-LOCA hydrogen recombiner system is designated as a Class 2 system (SWEC safety class) which is consistent with Group B Quality Standards as required by SRP 6.2.5, par. II; SRP 3.2.2; and Reg Guide 1.26. The system is also consistent with Seismic Category I in accordance with the classification requirements of Reg Guide 1.29, SRP 3.2.1, and SRP 6.2.5, par. II.

1.4 LIQUID AND GASEOUS RADIATION WASTE (RADWASTE) HANDLING SYSTEMS

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the liquid and gaseous radwaste handling system are:

<u>Standard Review Plans</u>

SRP 2.4.13, Rev. 0, November 1975, Accidental Releases of Liquid Effluents in Ground and Surface Waters

SRP 11.1, Rev. 2, July 1981, Source Terms

SRP 11.2, Rev. 2, July 1981, Liquid Waste Management Systems

SRP 11.3, Rev. 2, July 1981, Gaseous Waste Management Systems

SRP 12.1, Rev. 2, July 1981, Assuring the Occupational Radiation Exposures are as Low as Is Reasonably Achievable

SRP 12.2, Rev. 2, July 1981, Radiation Sources

SRP 15.7.2, Rev. 1, July 1981, Radioactive Liquid Waste System Leak or Failure (release to atmosphere) (deleted)

SRP 15.7.3, Rev. 2, July 1981, Postulated Radioactive Release Due to Liquid-Containing Tank Failures

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.143, Rev. 1, October 1979, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

1.4.1 Detailed Discussion

The portions of the liquid and gaseous waste systems within the auxiliary building were reviewed to the above documents and found to be in basic compliance. The following system modifications would be required to achieve full compliance.

North Anna Unit 3 is in compliance with equipment redundancy, subsystem interconnection, and reserve storage capacity requirements. All tanks and pumps are redundant and have interconnections between redundant pairs with the exception of the single, 25,000 gallon, regenerant chemical waste tank. Overflow can run between redundant tanks. Additional holdup capacity is provided by the fluid waste treating tank and the low level waste drain tanks. Ultimate overflow is to the auxiliary building sump except for the high level waste drain tanks, which overflow to the low level tanks.

The design would meet the requirements of Reg Guide 1.143. This includes the outdoor regenerant chemical evaporator test tank, which would be diked and provided with a pipe tunnel. Stainless steel liners capable of holding one tank volume would be provided around all tanks in the liquid waste system which may contain high levels of radwaste. This would insure that any tank rupture would be contained and would not contaminate other areas of the auxiliary building. All tanks are located in cubicles and are equipped with overflow lines, floor drains, and sample lines which are rerouted into the liquid waste system to prevent and collect spills. Liquid level monitors with alarms would be installed to indicate high and low levels on the waste disposal and solidification control boards.

Reg Guide 1.26 outlines Quality Groups A-D for nuclear power plant components using ASME and ANSI Codes as standards. The liquid radwaste and decontamination systems are designed to the ASME Boiler and Pressure Code, Section VIII, B31.1, and Quality Group D. The analagous ANS designation is Nonnuclear Safety Class (SWEC Class 4).

The gaseous and liquid waste components would be arranged in the auxiliary building to provide physical and electrical separation.

1.5 PENETRATION AREA

1.5.1 Detailed Discussion

The penetration area in the auxiliary building houses the reactor containment isolation valves and piping connecting the containment with the auxiliary building. Because both redundant safety-related lines and nonsafety lines are located in this area, the penetrations were rearranged to provide physical and electrical separation. A significant number of existing installed penetrations would have to be relocated. The impacts for penetration relocation and design rework are addressed in Section D.2.2.2.

1.6 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.5.1, Rev. 2, July 1981, Engineered Safety Feature Atmosphere Cleanup Systems

SRP 9.4.3, Rev. 2, July 1981, Auxiliary and Radwaste Area Ventilation System

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintainance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.140, Rev. 1, October 1979, Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

1.6.1 Detailed Discussion

The HVAC system for the auxiliary building was reviewed to the documents identified above and significant system modifications would be required to achieve compliance. A discussion of these modifications is given below.

The previous design of the HVAC system for the ESF equipment cooling in the auxiliary building consisted of two redundant exhaust fans and two common charcoal filter banks. The system provided ESF equipment cooling by drawing outside air (at outside ambient temperature) through the building and exhausting to the atmosphere.

The current HVAC system for ESF equipment cooling and exhaust air filtration in the auxiliary building consists of unit coolers, redundant makeup units, redundant charcoal filter banks and fans, and associated piping, ductwork, dampers, and instrumentation. Redundant water chillers are provided.

SRP 6.5.1 requires additional instrumentation for surveillance of ESF filter train operation.

The previous design of the HVAC system for cooling of nonsafety-related equipment consisted of two makeup units, two exhaust fans, and two station common charcoal filter banks in the auxiliary building. The current nonsafety-related HVAC system consists of two make-up units, one charcoal filter bank, one exhaust fan, and unit coolers in various nonsafety-related areas within the building.

The main reasons for the above design changes are:

- Reg Guide 1.52, Position C.2.f, requires that the volumetric flow rate through a single cleanup train be limited to 30,000 cfm or below.

- VEPCO has requested that maximum design ambient temperatures be increased to 107 FDB and 82 FWB.

The main reason for the design changes for nonsafety-related equipment cooling is that Reg Guide 1.140 requires that the volumetric flow rate through a single cleanup train be limited to 30,000 cfm or below.

The previous design, once-through ventilation, would require greater volumetric air flow in order to maintain an adequate equipment ambient temperature. With a greater air flow, unacceptably large air systems would result and multiple filters would be required to maintain flow through each filter train at below 30,000 cfm.

1.7 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

<u>Standard Review Plans</u>

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guide

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.7.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment) has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design, the auxiliary building fire protection system consisted of:

- Standpipe system with hose racks
- Low pressure CO_2 system for the electrical cable penetration areas
- Low pressure CO_2 system for the charcoal filter assemblies

The following modifications would be required to comply with the requirement of the reviewed documents:

- Automatic water spray system in lieu of the low pressure CO_2 system for the electrical cable penetrations areas
- Water spray system in lieu of the low pressure CO_2 system for all of the charcoal filter assemblies

1.8 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system (EFDS) are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.8.1 Detailed Discussion

The portion of the EFDS which exists in the auxiliary building was reviewed to the above documents and found to be in basic compliance. Systems modifications described in this section would be required to achieve full compliance.

The EFDS in the auxiliary building would require additional sumps and associated piping and pumps to support the rearrangement of auxiliary building equipment into separate ESF/nonsafety-related areas. Existing Unit 3 and Unit 4 pumps would be utilized. There is a significant impact as a result of the redesign of the auxiliary building to the separation criteria discussed in the individual system sections.

This is because the existing auxiliary building mat floor drain system is not compatible with the revised auxiliary building arrangements.

1.9 POST-ACCIDENT SAMPLING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the post-accident sampling system are:

Standard Review Plan

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.9.1 Detailed Discussion

Reg Guide 1.97, Table 2, and SRP 9.3.2 require an installed capability to sample the following systems/areas following an accident:

- Containment sump
- Containment atmosphere
- Reactor coolant system
- All emergency core cooling system pump sumps (including those in the safeguards, decay heat/ quench spray, and auxiliary buildings)

Onsite analysis capability over extended radioactivity and chemical concentrations is also required.

System modifications would be required to obtain samples from all the systems and areas identified above. In addition, modifications are necessary for handling and analyzing the required samples.

1.10 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and effluent sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.10.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the auxiliary building were reviewed to the above documents. The system modifications below would be required to achieve full compliance.

The auxiliary building ventilation exhaust no longer discharges to the common ventilation vent stack. SRP 11.5, Table 1A, Item 3 requires that continuous effluent radiation monitoring be provided for all individual building ventilation exhaust points. Therefore, gaseous and particulate radiation monitoring capability, with provisions for obtaining local grab samples, would be provided for the auxiliary building ventilation exhaust points. In addition, the auxiliary building ventilation exhaust, as an identified release point, would be monitored for noble gas activity and vent flow rate over extended accident ranges as required by Reg Guide 1.97, Table 2. Also, sampling capability for particulates and halogens, with onsite analysis, would be required.

1.11 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the radiation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.11.1 Detailed Discussion

Area radiation monitoring in the auxiliary building is discussed below. Post-accident, process, and effluent monitoring are covered in Sections E.1.9 and E.1.10.

The design of the area radiation monitoring system within the auxiliary building has not been finalized. However, the system would comply with the above documents.

Reg Guide 1.97, Table 2, requires that extended-range area radiation monitoring instrumentation be provided in the containment penetration areas of the auxiliary building.

E. AUXILIARY BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of the system changes in the auxiliary building are described in Section 1. The general changes to the structure are as follows.

2.1 IMPACT DUE TO LICENSING CHANGES

The arrangement of the auxiliary building has been completely revised to provide physical separation of redundant safety-related equipment. The overall building dimensions, height, floor elevations, and the locations and thickness of interior walls have changed. The only area of the building that reflects the original layout is the electrical tunnel and penetration area on the east side.

The provisions of SRP 3.8.4, Rev. 1, July 1981, Other Seismic Category I Structures, make the use of masonry walls technically unfeasible. Therefore, reinforced concrete walls with removable sections will be provided.

Impacts associated with the 3-D seismic requirements of Reg Guides 1.60, 1.61, and 1.92 are listed in the following section.

2.2 3-D SEISMIC IMPACTS

The auxiliary building has continuous reinforced concrete walls around the exterior at the mat level and 3 ft thick interior shear walls at the 25 7/8, 28 1/4, and J3 lines. These walls have been provided to resist higher seismic shear forces due to increased building weight and height.

F. FUEL BUILDING

SECTION 1

AREA SYSTEMS

1.1 FUEL POOL COOLING AND PURIFICATION SYSTEM (INCLUDING FUEL STORAGE AND HANDLING)

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fuel pool cooling and purification system are:

Standard Review Plans (SRP)

SRP 3.2.1, Rev. 0, November 1975, Seismic Classification

SRP 3.2.2, Rev. 0, November 1975, System Quality Group Classification

SRP 9.1.1, Rev. 2, July 1981, New Fuel Storage

SRP 9.1.2, Rev. 2, March 1979, Spent Fuel Storage

SRP 9.1.3, Rev. 1, July 1981, Spent Fuel Pool Cooling and Cleanup System

SRP 9.1.4, Rev. 2, July 1981, Light Load Handling System (related to refueling)

SRP 9.1.5, Rev. 0, July 1981, Overhead Heavy Load Handling Systems

Reg Guides

Reg Guide 1.13, Rev. 1, December 1975, Spent Fuel Storage Facility Design Basis (for comment)

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

1.1.1 Detailed Discussion

The fuel pool cooling and purification system, including fuel storage and handling systems, was reviewed to the above documents and several system modifications would be required to achieve full compliance.

These modifications are based on compliance with the above documents and the following increase in scope of the fuel storage capacity.

The fuel pool will have the capacity to store 4,400 fuel assemblies. This includes approximately 2,150 fuel assemblies from Unit 3 (40 yr storage) and 2,250 fuel assemblies from Units 1 and 2.

SRP 9.1.3, par. III, established the fuel pool cooling requirements. Based on the increased pool size and SRP 9.1.3 requirements, the existing fuel pool cooling heat exchangers and pumps would not be adequate for the long-term heat loads due to increased fuel storage. Therefore, additional heat exchangers and pumps would be required.

Extensive rearrangement of the building is required to facilitate the increased pool capacity and the physical separation of the safety-related redundant cooling trains.

Existing fuel handling equipment, such as cranes, will be utilized where design permits and where practical. All fuel handling operations within the fuel building will be in accordance with the SRP 9.1.2 and SRP 9.1.5. To maximize space requirements for storing the increased number of fuel assemblies, poison fuel racks will be used. The fuel rack design will be in accordance with SRP 9.1.1 and SRP 9.1.2.

The building ventilation currently is designed on the assumption that the cladding of 56 fuel pins is breached during a fuel handling accident (PSAR Section 14.2.2.3). Reg Guide 1.13 requires that the design be based on the assumption that 208 pins in one assembly are breached. Radiation release calculations need to be redone based on 208 pins. Equipment redesign would be specified should the offsite doses be found unacceptable.

1.2 HVAC SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 6.5.1, Rev. 2, July 1981, Engineered Safety Feature Atmosphere Cleanup Systems

SRP 9.4.2, Rev. 2, July 1981, Spent Fuel Pool Area Ventilation System

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado for Nuclear Power Plants

1.2.1 Detailed Discussion

The HVAC system for the fuel building was reviewed to the above documents and significant system modifications would be required to achieve compliance. A discussion of these modificatons is given below.

The previous design of the HVAC system in the fuel building consisted of one fuel pool air supply unit, two 50 percent exhaust fans with provision for diversion through a common station charcoal filter assembly located in the auxiliary building, and one pump room air supply unit.

The current HVAC system in the fuel building consists of two 50 percent chilled water cooling air supply units and two 50 percent exhaust fans with provision to divert exhaust air through redundant QA Category I fans and charcoal filter trains during refueling and following a postulated fuel handling accident. The fuel pool cooling equipment cubicles are cooled by Category I chilled water unit coolers during all modes of plant operation.

The main reasons for the design changes outlined above are as follows:

- Reg Guide 1.52 position C.2.f requires that the volumetric flow rate through a single cleanup train be limited to 30,000 cfm or below.

- VEPCO has requested that the maximum design ambient temperature be increased to 107 FDB and 82 FWB.

With the higher temperature, the previous design (once-through cooling) would require greater volumetric air flow in order to maintain an adequate room temperature. With a greater air flow, unacceptably large systems would result and multiple filters would be required to maintain flow through filter trains at below 30,000 cfm.

SRP 6.5.1 requires additional instrumentation for surveillance of the filter drain operation.

1.3 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guide

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.3.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of SRP 9.5.1 and BTP CMEB 9.5-1. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The previous design of the fuel building fire protection system provided a standpipe system with hose stations.

To comply with the requirements of Reg Guide 1.52, a water spray system would be required for the two new safety-related charcoal filter assemblies.

An automatic fire suppression system would be required for the cask wash down area to comply with the requirements of BTP CMEB 9.5-1.

1.4 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system (EFDS) are:

<u>Standard Review Plans</u>

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

<u>Reg Guides</u>

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.4.1 Detailed Discussion

The EFDS in the fuel building will require redesign of the floor and equipment drainage to support the fuel building rearrangement. No significant impact results from the redesign of the drainage system.

1.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and post-accident sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.5.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems, which exist in the fuel building, were reviewed to the above documents. System modifications described below would be required to achieve full compliance.

The fuel building ventilation exhaust no longer discharges to the common ventilation vent stack. SRP 11.5, Table 1A, Item 6, requires that continuous effluent radiation monitoring be provided for the fuel storage areas ventilation system exhaust. Therefore, gaseous and particulate radiation monitoring capability, with provisions for obtaining local grab samples, would be required.

In addition, SRP 11.5, Table 1B, Item 5, requires that continuous process radiation monitoring be provided for the refueling/spent fuel pool purification system. Therefore, process radiation monitoring would be required for this system.

The impacts of the above system modifications are presented in the project position statement for Reg Guide 1.97.

1.6 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the area radation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guide

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.6.1 Detailed Discussion

The design of the area radiation monitoring system within the fuel building will comply with the documents listed above.

F. FUEL BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of the system changes in the fuel building, due to licensing issues, are in Section 1. The general changes to the structure are as follows.

2.1 IMPACT DUE TO LICENSING CHANGES

Due to separation requirements for safety-related equipment, the size of the fuel building has been significantly increased to accommodate the increased storage capacity of the pool for Units 1, 2, and 3.

The provisions of SRP 3.8.4, Rev. 1, July 1981, Other Seismic Category I Structures, make the use of masonry walls technically unfeasible. Reinforced concrete walls with removable sections will be utilized and will also provide separation between safety-related equipment.

2.2 3-D SEISMIC IMPACTS

The fuel building is a heavily reinforced concrete building. The seismic design requirements of Reg Guides 1.60, 1.61, and 1.92 increase the analysis and design requirements for the structure.

G. CONTROL BUILDING

SECTION 1

AREA SYSTEMS

1.1 CONTROL SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the control system are:

<u>Standard Review Plans (SRP)</u>

SRP 3.5.1.1, Rev. 2, July 1981, Internally Generated Missiles (outside containment)

SRP 3.5.1.3, Rev. 2, July 1981, Turbine Missiles

SRP 6.4, Rev. 2, July 1981, Control Room Habitability Systems

SRP 6.5.1, Rev. 2, July 1981, ESF Atmosphere Cleanup Systems

SRP 7.1, Rev. 2, July 1981, Instrumentation and Controls - Introduction

SRP 7.2, Rev. 2, July 1981, Reactor Trip System

SRP 7.3, Rev. 2, July 1981, Engineered Safety Features Systems

SRP 7.4, Rev. 2, July 1981, Safe Shutdown Systems

SRP 7.5, Rev. 2, July 1981, Information Systems Important to Safety

SRP 7.6, Rev. 2, July 1981, Interlock Systems Important to Safety

SRP 7.7, Rev. 2, July 1981, Control Systems

SRP 9.4.1, Rev. 2, July 1981, Control Room Area Ventilation System

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

SRP 18.0, Rev. 0, July 1981, Human Factors Engineering/Standard Review Plan Development

Reg Guides

Reg Guide 1.22, Rev. 0, February 1972, Periodic Testing of Protection System Actuation Functions (Safety Guide 22)

Reg Guide 1.47, Rev. 0, May 1973, Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Reg Guide 1.52, Rev. 2, March 1978, Design, Testing, and Maintenance Criteria for Post Accident Engineering-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.53, Rev. 0, June 1973, Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

Reg Guide 1.62, Rev. 0, October 1973, Manual Initiation of Protective Actions

Reg Guide 1.68.2, Rev. 1, July 1978, Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear P[illegible] Plants to Assess Plant Conditions during and following a[illegible]ident

Reg Guide 1.115, Rev. 1, July 1977, Protection against Low-Trajectory Turbine Missiles

Reg Guide 1.139, Rev. 1 (draft 2), March 1980, Guidance for Residual Heat Removel to Achieve and Maintain Cold Shutdown (for comment)

1.1.1 Detailed Discussion

A number of concerns led to the evolution of a separate control building. They are discussed in the following sections.

The review of the existing service building was conducted following the cancellation of Unit 4. Initial concerns focused on the spatial relationship between the control room, turbine, and main steam lines. Relocation of the control building reduced exposure to main steam line pipe break, turbine missile, and noise concerns.

However, the station layout would permit only a long narrow control room because of the required separation between cable spreading rooms, relay rooms, and an auxiliary shutdown room.

Concerns with electrical separation, Reg Guide 1.75, fire protection, and HVAC, as well as the post-TMI regulatory requirements identified in Reg Guide 1.97, Reg Guide 1.139, and SRP 18.0 compounded space problems in the service building. These competing regulatory requirements imposed demands for space which contributed to the separation of the control complex and the service building.

The major items requiring the space are:

1. Separation of safety and nonsafety cable spreading rooms, relay rooms, batteries, and inverters

2. Ventilation system upgrade to separate, redundant systems

3. Expanded capability of auxiliary shutdown room functions

4. Addition of safety-related relays as a consequence of licensing changes to SWEC and B&W fluid systems

5. TMI requirements of increased computer capability, which also necessitates additional ventilation

6. Control room space requirements increased to address human factors concerns and the additional information processing and display required by TMI

7. Fire protection

1.2 ELECTRICAL SEPARATION

Compliance with the requirements of Reg Guide 1.75 and BTP CMEB 9.5-1 has direct impacts on the control building size and the control room layout. For example, separation of redundant, safety-related circuits from nonsafety-related circuits by a fire wall requires:

1. Safety-related relay rooms that are completely separated from nonsafety-related relay rooms

2. Redundant relay rooms (by train) that are separated by a fire wall

3. Separate spreading areas for both safety-related cables and nonsafety-related cables

4. Separation of battery and inverter rooms

5. Separation of auxiliary shutdown room from within a single emergency switchgear room

1.3 FIRE PROTECTION

Reg Guide 1.120 is still on review and has not been formally issued. BTP CMEB 9.5-1 (NUREG 0800) is to be used as a basis for evaluation per NRC directive until Reg Guide 1.120 is issued.

The control building safety-related areas are arranged to provide separation for redundant safety-related systems and equipment so that both are not subject to damage from a single fire hazard. This is accomplished by placing each redundant component in a separate fire area.

In compliance with the requirements of the reviewed documents, an automatic water spray system has replaced the low pressure CO_2 system for the cable spreading areas and tunnels.

1.4 VENTILATION

The previous design of the control room pressurization system consisted of two redundant sets of compressed bottled air for the first hour following an accident, and outside air filtered by a single charcoal filter and fan assembly following the depletion of bottled air.

The current design of the control room pressurization system consists of two redundant charcoal filter and fan units supplying outside air from either of two independent, missile-protected, well-separated air intakes. In addition, a 6 hour breathing air supply will be available during an isolated condition with all intakes closed. During accidental releases of radioactive gases, the pressure in the control room will be maintained at approximately 1/8 inch.

The main reasons for the design changes are based on SRP 6.4 and Reg Guide 1.78. These documents describe assumptions acceptable to the NRC staff and are to be used in assessing the habitability of the control room during and after a postulated external release of toxic or radioactive gases.

The major physical requirements for the redundant control room filters have been considered and form the basis of the space allotted. These include air flow rate limits, space for filter removal, and shielding. In addition to fire protection requirements and Reg Guide 1.52, SRPs 6.5.1 and 9.4.1 form a major basis of the ventilation system equipment layout and space considerations to provide the required redundancy, separation, and single active failure protection.

1.5 SAFE SHUTDOWN

Space has been provided for compliance with both SRP 7.4 and Reg Guide 1.139. Hot shutdown is currently designed for, and cold shutdown is coordinated through, the auxiliary shutdown room and accomplished from remote control stations, such as emergency switchgear and/or motor control centers.

1.6 FLOODING

Safety-related electrical, instrumentation, control, and mechanical equipment must be protected from flooding. This flooding could originate from either natural environmental phenomena or passive failure of piping systems inside the control building. All safety-related electrical equipment has been located above El 271 ft, which is grade elevation.

The control building layout reduces the internal flooding potential by routing HVAC chilled water lines in isolated pipe chases and eliminating process piping within the electrical and control areas of the building.

1.7 TURBINE MISSILES

The criteria for evaluation of low trajectory turbine missiles is provided in Reg Guide 1.115 and SRP 3.5.1.3. The location of the control building with regard to the turbine reduces the probability of missile impact.

1.8 STEAM LINE BREAK AND CRACKS

The pipe rupture evaluation is performed in accordance with SRP 3.6.1, which requires that all high energy lines outside the containment be reviewed for the effect of breaks and cracks. The new location of the control building eliminates any problems associated with postulated main steam or feedwater pipe rupture.

1.9 CONTROL ROOM HAZARDS AND HABITABILITY

SRP 6.4 establishes guidelines for evaluation of control room habitability after the accidental release of toxic or radioactive gases and for the long term after a design basis radiological release.

Due to the possibility of higher design radiological releases associated with the TMI accident analysis and to concerns with toxic releases, two independent air intakes for the control room are included in the evaluation. Dual remote air intakes for the control room will provide a lower probability that both would be contaminated from the same source. Gas, smoke, and radiation detectors and isolation valves (dampers) will be provided in each outside air intake for the control room.

SRP 6.5.1 requires additional instrumentation for surveillance of filter train operation.

Past experience has shown problems with both the accountability of pressure boundary penetrations and the sealing of doors through the pressure boundary. SWEC's intent is to minimize the extent of the area to be pressurized and the number of doors and penetrations consistent with other requirements.

The Unit 3 pressure boundary will include the control room, HVAC equipment room, and the safety-related relay rooms.

The first hour air bottle system has been replaced by a redundant fan and filter system in combination with two independent air intakes. This is consistent with SRP 6.4.

1.10 COMPLIANCE WITH TMI CONCERNS

1.10.1 NUREG 0578 - Control Room

The control building design will accommodate the requirements of NUREG 0578 for information and display.

The four vertical breakfront control board design accommodates an improved CRT capability, better functionality, and an acceptable operator interface. The distributed control board concept also allows for easier compliance with separation criteria.

The control building is presently being designed as a conventional hard wired point-to-point terminated system incorporating a human engineered operator interface.

The EOF would be located near the site.

Requirements which have dictated a new computer system include system redundancy, availability, storage of post-accident data, and additional capacity to drive the distributed control board CRT displays.

1.10.2 NUREG 0660-Control Room Human Engineering and Design

1.10.2.1 Control Area Noise

The control areas (main control room, safety-related relay room, computer room, nonsafety-related relay room, and auxiliary shutdown room) are by design and nature quiet (nonmachinery) areas. The majority of electronics and electro-mechanical device enclosures are designed for external cooling. There are no internal fans. Various room ventilation system ambient noise levels will be kept to a minimum.

1.10.2.2 Safety System Status Monitoring (Revised Reg Guide 1.47)

The previous main control board design was in conformance with Reg Guide 1.47. However, a revision to the Reg Guide is expected and should be accommodated on the new main control board array.

G. CONTROL BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of the system changes in the control building are in Section 1. The general changes to the structures are as follows:

The control room and associated facilities were located in the service building, but have been relocated west to the control building. The new building is approximately 162 ft long x 172 ft wide and has four floors.

2.1 TOTAL BUILDING IMPACT DUE TO ALL LICENSING CHANGES

Relocation of the control building and changes in its size are due primarily to separation, fire protection, and ventilation system upgrading. Also, building changes were introduced as a consequence of added safety-related equipment and TMI-related concerns affecting the computer system, displays, and human factors engineering.

2.2 3-D SEISMIC IMPACTS

The control building will be designed for 3-D seismic requirements per Reg Guides 1.60, 1.61, and 1.92.

H. SERVICE BUILDING

SECTION 1

AREA SYSTEMS

There are no safety-related systems located in the service building. A nonsafety-related Class 4 portion of the steam generator blowdown system is routed through the service building and is discussed in Section C.1.4. A fire protection system is also provided and is discussed in Section H.1.2.

The potable and sanitary water system in the service building complies with the requirements of SRP 9.2.4, Rev. 2, July 1981, Potable and Sanitary Water Systems, with no associated impact. The system has no interconnections with any other system having the potential for containing radioactive material.

Changes to the service building, described in Section H.2 and the project position on Regulatory Guide 1.75, would require modifications to various systems in the service building.

The Technical Support Center (TSC) will be located on the upper elevations of the west end of the service building and will comply with the requirements of NUREG 0696, Functional Criteria for Emergency Response Facilities. Impacts due to the addition of the TSC include:

1. Addition of a TSC habitability and radiation monitoring system, as required by NUREG 0696, Section 2.6;

2. Addition of safety parameter display system processing and readout equipment and a portion of the main plant computer system with associated readout equipment;

3. Structural-related impacts due to 3-D seismic requirements are presented in Section H.2.

1.1 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

<u>Standard Review Plans</u>

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Req Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.1.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of SRP 9.5.1 and BTP CMEB 9.5-1. Reg Guide 1.120, Rev. 1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design, the service building fire protection system included a standpipe system with hose stations and a low pressure CO_2 system for the cable spreading areas and tunnels.

Consistent with the requirements of SRP 9.5.1 and BTP CMEB 9.5-1, automatic water spray systems have been provided in the current design in place of the CO_2 systems for the cable spreading areas and tunnels. Fire protection for the TSC will be provided by hose stations.

H. SERVICE BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The general changes to the structures are as follows.

The old service building included the control room and condensate polishing, health physics, and normal switchgear areas. The new service building contains the health physics area, the normal switchgear area, and the technical support center.

2.1 TOTAL BUILDING IMPACT DUE TO ALL LICENSING CHANGES

The relocation of the control room has a major impact on the service building. The condensate polishing building has been relocated because of space requirements for cable tunnels from the service building to the control building. The technical support center has been added to comply with NUREG 0696.

2.2 3-D SEISMIC IMPACTS

The service building would have to be redesigned for the 3-D seismic requirements of Reg Guides 1.60, 1.61, and 1.92.

I. SERVICE WATER PUMPHOUSE

SECTION 1

AREA SYSTEMS

1.1 SERVICE WATER SYSTEM

Those documents reviewed which provide guidance and/or established criteria for the design basis of the service water system are:

Standard Review Plans (SRP)

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.2.1, Rev. 2, July 1981, Station Service Water System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.1.1 Detailed Discussion

That portion of the service water system which exists in the service water pumphouse (SWPH) was reviewed to the above documents. System modifications described in this section would be required to achieve full compliance.

The service water pumps, traveling water screens, and screen wash pumps are located in the SWPH. Due to increased service water flow requirements resulting from revision of the component cooling water system, increased flow for air conditioning units, and use of water-cooled diesels, the existing capacity of the service water pumps will not be adequate. Additional pumps would be required.

Provision of adequate physical and electrical separation will require extensive modifications to the existing portion of the SWPH. Two 50 percent pumps will be provided in each intake bay.

The service water system is designated a Class 3 system (SWEC safety class), which is consistent with Group C Quality Standards as required by SRP 9.2.1, par. II.7; SRP 3.2.2; and, Reg

Guide 1.26. These portions of the system are also designated Seismic Category I in accordance with the classification requirements of SRP 9.2.1, par. II.8; SRP 3.2.1; and, Reg Guide 1.29.

Confirmation that the postulated single failure (active or passive) will not result in a loss-of-function would be documented in the system's failure modes and effects analysis report required by SRP 9.2.1, Section III.2.

1.2 HVAC SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation System

Reg Guides

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado for Nuclear Power Plants

1.2.1 Detailed Discussion

The HVAC system for the SWPH was reviewed to the above documents and no system modifications are required, with the exception of adding tornado protection air duct dampers. However, due to the VEPCO request to increase maximum design temperature to 107 FDB and 82 FWB, and to the increased size of the service water pump motors, the previously designed ventilation system will be enlarged.

1.3 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Programs for Nuclear Power Facilities Operating Prior to January 1, 1979

1.3.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The previous design of the SWPH fire protection system, which consisted of hoses attached to the pump test manifold on the exterior of the building, complies with the requirements of the reviewed documents. No modifications of the fire protection system are required.

1.4 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the equipment and floor drainage system (EFDS) are:

Standard Review Plans

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.3.3, Rev. 2, July 1981, Equipment and Floor Drainage System

Reg Guides

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for comment)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

1.4.1 Detailed Discussion

The portion of the EFDS which exists in the SWPH was reviewed to the above documents and found to be in compliance.

1.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and post-accident sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

1.5.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the SWPH were reviewed to the above documents and the system modification described below would be required to achieve full compliance.

SRP 11.5, Table 1B, Item 3, requires that service water effluent be monitored continuously. Although this requirement was previously met, one additional radiation monitor would be required as a result of service water piping redesign.

I. SERVICE WATER PUMPHOUSE

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of the system changes in the service water pumphouse are in Section 1. The general changes to the structures are as follows.

2.1 IMPACT DUE TO LICENSING CHANGES

The existing structure requires revision to El 326 ft with additional walls above. All modes of primary service water system operation, including the emergency core cooling system, will be on the reservoir.

2.2 3-D SEISMIC IMPACTS

The service water pumphouse will be designed in agreement with Reg Guides 1.60, 1.61, 1.92, and 1.122 and SRPs 3.7.1 and 3.7.2.

J. WASTE STORAGE BUILDING

SECTION 1

AREA SYSTEMS

1.1 WASTE SOLIDIFICATION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the waste solidification system (WSS) are:

Standard Review Pl n (SRP)

SRP 11.4, Rev. 2, July 1981, Solid Waste Management Systems

Reg Guide

Reg Guide 1.143, Rev.1, October 1979, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

Other Related Documents

BTP ETSB 11-3, Rev.2, July 1981, Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants

1.1.1 Detailed Discussion

The WSS was reviewed to the above documents. Modifications would be required to achieve full compliance.

The current urea formaldehyde system was reviewed to the requirements of SRP 11.4 and BTP ETSB 11-3 and it is unlikely the system can produce a water- free product to meet the requirements of BTP ETSB 11-3. The following solid waste systems are being considered.

- Provisions for installation of solidification and volume reduction to meet potential needs to minimize shipments and storage of radioactive waste.
- Use of high integrity containers to ship dewatered resins.

In order to comply with required solidification process control pr s, a means of adjusting waste parameter, such as waste pH, wil be provided. A sampling system for all waste to be sol ified would be required.

The design of the solid waste building (nonseismic, QA Category III) would comply with supporting documents in Reg Guide 1.143.

1.2 HVAC SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC systems are:

Reg Guide

Reg Guide 1.140, Rev. 1, October 1979, Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

1.2.1 Detailed Discussion

The HVAC system for the waste solidification building was reviewed to the above document and some system modifications are required to achieve compliance. These modifications are discussed below.

The previous design of the HVAC system for the waste solidification building consisted of one waste solidification building air supply unit, one exhaust high efficiency particulate adsorbtion (HEPA) filter train, and two 50 percent capacity exhaust fans. The current HVAC system consists of two 50 percent capacity chilled water cooling air supply units, one exhaust air HEPA filter train, and two 50 percent capacity exhaust fans.

The above design changes are primarily due to the Reg Guide 1.140 requirements which limit the volumetric flow rate through a single cleanup train to 30,000 cfm or below.

With the higher temperature, the previous design (once-through cooling) would require a greater volumetric air flow to maintain an adequate room temperature. With a greater air flow, unacceptably large systems would result and multiple filters would be required to maintain flow through filter trains at below 30,000 cfm.

1.3 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.3.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been extensively revised and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The previous design of the waste solidification building fire protection system consisted of a standpipe system with hose stations. An automatic fire suppression system would be required to comply with BTP CMEB 9.5-1.

1.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design of the process and post-accident sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

1.4.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the waste solidification building were reviewed to the above documents and found to be in basic compliance.

J. WASTE STORAGE BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of system changes in the waste storage building are in Section 1. The general changes to the structure are as follows.

The original waste storage building was a reinforced concrete structure below grade and a structural steel frame with metal siding above grade. The revised building is a reinforced concrete structure both above and below grade.

2.1 IMPACT DUE TO LICENSING CHANGES

Reg Guide 1.143, Rev.1, October 1979, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, and SRP 11.4, Rev. 2, July 1981, Solid Waste Management Systems, require that the structure be designed to resist seismic effects and contain any spills within the building.

2.2 3-D SEISMIC IMPACTS

The waste storage building is a reinforced concrete structure founded on soil. The seismic design requirements of Reg Guides 1.60, 1.61, and 1.92 increase the analysis and design requirements for the structure.

K. TURBINE BUILDING

SECTION 1

AREA SYSTEMS

1.1 TURBINE GENERATOR SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the turbine generator system and associated systems are:

<u>Standard Review Plans (SRP)</u>

SRP 10.2, Rev. 2, July 1981, Turbine Generator

SRP 10.4.1, Rev. 2, July 1981, Main Condensers

SRP 10.4.2, Rev. 2, July 1981, Main Condenser Evacuation System

SRP 10.4.3, Rev. 2, July 1981, Turbine Gland Sealing System

SRP 10.4.4, Rev. 2, July 1981, Turbine Bypass System

SRP 10.4.5, Rev. 2, July 1981, Circulating Water System

SRP 10.4.6, Rev. 2, July 1981, Condensate Cleanup System

<u>Reg Guides</u>

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Reg Guide 8.8, Rev. 4 (draft), March 1979, Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.1.1 Detailed Discussion

The turbine generator system and associated systems located in the turbine building and condensate polishing building were reviewed to the above documents and found to be in basic compliance. Several system modifications would be required to achieve full compliance.

Hazards evaluations and failure modes and effects analyses of the systems in the turbine building would be performed to verify that:

- Failure of a main condenser and the resulting flooding will not preclude operation of any essential systems, as required by SRP 10.4.1, par. III.3.a.
- Isolation of the main steam system can be achieved on loss of condenser vacuum, as required by SRP 10.4.1, par. III.3.b.
- Failure of the turbine bypass system to operate will not preclude operation of any essential systems, as required by SRP 10.4.4, par. III.3.a.
- Failure of the turbine bypass system high energy piping will not have adverse effects on any safety-related systems or components that may be located close to the system, as required by SRP 10.4.4, par. III.3.b.

Structural modifications of the condensate polishing building to achieve ALARA, as required by Reg Guide 8.8 and SRP 10.4.6, would require rearrangement of portions of the condensate cleanup system.

1.2 VENTILATION SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the ventilation system are:

Standard Review Plan

SRP 9.4.4, Rev. 2, July 1981, Turbine Area Ventilation System

Reg Guides

None

1.2.1 Detailed Discussion

The ventilation system for the turbine building was reviewed to the above document.

The turbine building does not house any safety-related equipment. Therefore, the ventilation system is not safety-related and SRP 9.4.4 is not applicable for the turbine building.

1.3 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.3.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The previous design of the turbine building fire protection system consisted of the following:

1. A standpipe system with hose racks was located throughout the building.

2. Automatic sprinkler systems were provided under the mezzanine and operating levels, in the turbine oil purifier room, and in the turbine oil storage room.

3. Water spray systems were provided for the hydrogen seal oil unit and for the turbine oil rese[illegible] and cooler zones.

4. A low pressure CO_2 system was provided for turbine bearing areas.

All of the above fire protection systems are intended to ensure the integrity of the fire barrier which separates the turbine building from the adjacent structures. This is in compliance with the reviewed documents.

1.4 AREA RADIATION MONITORING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the radation monitoring system are:

Standard Review Plan

SRP 12.3-12.4, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guides

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

Reg Guide 8.8, Rev. 3, June 1978, Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable

1.4.1 Detailed Discussion

The radiation monitoring system in the turbine building and condensate polishing building includes only area radiation monitoring. Process and effluent monitoring is discussed in Section K.1.5. Post-accident sampling requirements do not apply to the turbine building or the condensate polishing building.

The design of the area radiation monitoring system in the turbine building and the condensate polishing building will comply with the above documents.

1.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The documents reviewed which provide guidance and/or establish criteria for the design of the process and effluent radiological monitoring and sampling systems are:

Standard Review Plans

SRP 9.3.2, Rev. 2, July 1981, Process and Post-Accident Sampling Systems

SRP 11.5, Rev. 3, July 1981, Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

Reg Guide

Reg Guide 1.97, Rev. 2, December 1980, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions during and following an Accident

1.5.1 Detailed Discussion

The portions of the process and effluent radiological monitoring and sampling systems which exist in the turbine building and condensate polishing building were reviewed to the above documents and found to be in compliance.

1.6 MAIN STEAM SUPPLY SYSTEM

Although portions of the main steam system are located in the turbine building, the system is addressed in Section C.1.1.

K. TURBINE BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of the system changes in the turbine building are in Section 1. The general changes to the structure are as follows.

The length of the turbine building has been extended for three more bays to allow for installation and removal of the rotor and stator. This space was allocated previously to the Unit 4 turbine building. This results in a revision to the foundation and roof framing for wind loading. The steel framing and foundations also have major impacts from licensing changes.

The turbine building foundations, horizontal wind bracing, and vertical bracing for wind require redesign based on the extended length.

2.1 TOTAL BUILDING IMPACT DUE TO LICENSING CHANGES

Portions of turbine building structural steel framing and foundations are designed to comply with SRP 3.5.1.4, Rev. 2, July 1981, Missiles Generated by Natural Phenomena, and Reg Guide 1.117, Rev. 1, April 1978, Tornado Design Classification. The C-line (wall between the turbine building and the service building) steel framing, including the outriggers, will be designed to prevent a collapse of the turbine building onto the service building.

The C-line steel columns are designed for the combination of tornado missiles and wind loadings. An outrigger system is included on the roof of the service building utilizing a redundant framing system so that a failure of any one member would not result in the collapse of the turbine building onto the service building.

The service building changes affected the outrigger configuration.

2.2 3-D SEISMIC IMPACTS

The turbine building will be designed per the requirements of SRP 3.7.2, Rev. 1, July 1981, Seismic System Analysis, so that it will not collapse on the top of the service building during an earthquake. The foundation and structural steel of the C-line are required to comply with Reg Guides 1.60, 1.61, and 1.92 because the C-line is common to the service building.

L. INTAKE STRUCTURE

SECTION 1

AREA SYSTEMS

1.1 CIRCULATING WATER SYSTEM

The circulating water system components located at the intake structure have not been altered due to Reg Guide or Standard Review Plan (SRP) requirements.

1.2 SCREENWASH SYSTEM

The screenwash system has not changed due to Reg Guide or SRP requirements.

1.3 RAW WATER SYSTEM

The raw water system components located within the intake structure have not changed due to Reg Guide or SRP requirements.

1.4 SERVICE WATER SYSTEM

The service water system components located within the intake structure have not been altered due to Reg Guide or SRP requirements.

1.5 HVAC SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the HVAC system are:

Standard Review Plans

SRP 3.2.1, Rev. 0, November 1975, Seismic Classification

SRP 3.2.2, Rev. 0, November 1975, System Quality Group Classification

SRP 9.4.5, Rev. 1, March 1978, Engineered Safety Feature Ventilation System

Reg Guides

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado for Nuclear Power Plants

1.5.1 Detailed Discussion

The HVAC system for the intake structure was reviewed to the above documents and the following system modification would be required as a result of the increase in maximum design outdoor temperature to 107 FDB and 82 FWB. The previously designed ventilation system is replaced with a direct expansion air conditioning system. This will maintain a room temperature no higher than the 104 F which is suitable for the existing pump motors.

1.6 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

<u>Standard Review Plans</u>

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

<u>Reg Guides</u>

None

<u>Other Related Documents</u>

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.6.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev. 1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design, the fire protection system of the intake structure consisted of yard hydrants located nearby. No modifications of the design are required for compliance with the reviewed documents.

L. INTAKE STRUCTURE

SECTION 2

BUILDING IMPACT ASSESSMENT

2.1 STRUCTURAL RELATED ISSUES

The details of system changes are in Section 1. The general changes to the structure are as follows.

The entire circulating water (CW) intake structure will be constructed, although Unit 4 is deleted. Extra space will be used by facilities for air conditioning equipment required when ambient temperature approaches 107 F. A structural steel framed enclosure with insulated siding will be added. The concrete floor slab at El 265 will require reanalysis due to an increase in the high water level.

The CW discharge tunnel will be investigated for seismic capability and for modifications due to the introduction of the concrete drainage channel into the Unit 4 discharge tunnel.

2.2 TOTAL BUILDING IMPACT DUE TO SYSTEM LICENSING CHANGES

There is no building impact due to system licensing changes for the intake structure or the CW discharge tunnel at the present time. The modifications discussed above reflect a change in design criteria.

2.3 3-D SEISMIC IMPACTS

The CW intake structure and the CW discharge tunnel will be redesigned for 3-D seismic requirements of Reg Guides 1.60, 1.61, and 1.92.

Since the service water piping and the concrete drainage channel discharge to the CW discharge tunnel, the tunnel will be investigated for seismic adequacy.

M. DIESEL GENERATOR BUILDING

SECTION 1

AREA SYSTEMS

Those documents reviewed which provide guidance and/or establish design criteria for systems or components associated with the diesel generator building are:

Standard Review Plans (SRP)

SRP 3.2.1, Rev. 0, November 1975, Seismic Classification

SRP 3.2.2, Rev. 0, November 1975, System Quality Group Classification

SRP 8.3.1, Rev. 2, July 1981, AC Power Systems (onsite)

SRP 8.3.2, Rev. 2, July 1981, DC Power Systems (onsite)

SRP 9.4.5, Rev. 1, March 1978, Engineered Safety Feature Ventilation System

SRP 9.5.1, Rev. 2, March 1978, Fire Protection Program

SRP 9.5.4, Rev. 2, July 1981, Emergency Diesel Engine Fuel Oil Storage and Transfer System

SRP 9.5.5, Rev. 2, July 1981, Emergency Diesel Engine Cooling Water System

SRP 9.5.6, Rev. 2, July 1981, Emergency Diesel Engine Starting System

SRP 9.5.7, Rev. 2, July 1981, Emergency Diesel Engine Lubrication System

SRP 9.5.8, Rev. 2, July 1981, Emergency Diesel Engine Combustion Air Intake and Exhaust System

Reg Guides

Reg Guide 1.6, Rev. 0, March 1971, Independence between Redundant Standby (onsite) Power Sources and between Their Distribution Systems (Safety Guide 6)

Reg Guide 1.9, Rev. 2, December 1979, Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (for comment)

Reg Guide 1.32, Rev. 2, February 1977, Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

Reg Guide 1.108, Rev. 1, August 1977, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

Reg Guide 1.137, Rev. 1, October 1979, Fuel-Oil Systems for Standby Diesel Generators

1.1 DIESEL GENERATOR FUEL OIL SYSTEM

Those documents reviewed which provide guidance and/or establish design criteria for the diesel generator fuel oil system are:

Standard Review Plan

SRP 9.5.4, Rev. 2, July 1981, Emergency Diesel Engine Fuel Oil Storage and Transfer System

Reg Guide

Reg Guide 1.137, Rev. 1, October 1979, Fuel-Oil Systems for Standby Diesel Generators

1.1.1 Detailed Discussion

In order to achieve compliance with Reg Guide 1.137 and ANSI Standard N-195, Fuel Oil Systems for Standby Diesel Generators, protection against external corrosion to the fuel oil system would be required in the form of a protective coating and an impressed current-type cathodic protection system.

1.2 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.2.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev.1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

In the previous design, the diesel generator building fire protection system consisted of a low pressure CO_2 system provided for the diesel generator room and the fuel oil tank room.

For compliance with the reviewed documents, an automatic sprinkler system would be required for the above areas in lieu of the CO_2 system. The backup for this fire protection system would be provided by the yard hydrants located in the proximity of the building.

1.3 HVAC System

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the diesel generator building HVAC system are:

Standard Review Plan

SRP 3.2.1, Rev. 1, July 1981, Seismic Classification

SRP 3.2.2, Rev. 1, July 1981, System Quality Group Classification

SRP 9.4.5, Rev. 2, July 1981, Engineered Safety Feature Ventilation Systems

Reg Guides

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado for Nuclear Power Plants

Reg Guide 1.117, Rev. 1, April 1978, Tornado Design Classification

1.3.1 Detailed Discussion

The HVAC system for the diesel generator building was reviewed to the above documents. The only system modification required would be the addition of tornado dampers due to the requirements of Reg Guides 1.76 and 1.117.

1.4 EMERGENCY DIESEL ENGINE COOLING WATER SYSTEM

Those documents reviewed which provide guidance and/or establish design criteria for the design basis of the diesel cooling water system are:

Standard Review Plans

SRP 9.2.1, Rev. 2, July 1981, Station Service Water System

SRP 9.5.5, Rev. 2, July 1981, Emergency Diesel Engine Cooling Water System

Reg Guides

Reg Guide 1.115, Rev. 1, July 1977, Protection against Low-Trajectory Turbine Missiles

Reg Guide 1.9, Rev. 2, December 1979, Selection, Design, and Qualification of Diesel Generator Units Used as (onsite) Electric Power Systems at Nuclear Power Plants (for comment)

1.4.1 Detailed Discussion

Each emergency diesel engine cooling water system (EDECWS) is protected from the effects of pipe break. Only the high or moderate energy piping in any individual diesel generator cell (the air starting system and associated piping dedicated to that particular diesel generator cell) will be seismically qualified and restrained to prevent damage to the EDECWS.

Each EDECWS meets the Reg Guide 1.9 Position 7 requirements related to engine cooling water protective interlocks, because temperature and pressure trips in the cooling water system will be by-passed during accident conditions.

1.5 EMERGENCY DIESEL ENGINE STARTING SYSTEM

Those documents reviewed which provide guidance and/or establish design criteria for the diesel engine starting systems are:

Standard Review Plan

SRP 9.5.6, Rev. 2, July 1981, Emergency Diesel Engine Starting System

Reg Guide

Reg Guide 1.9, Rev. 2, December 1979, Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (for comment)

Other Related Documents

NUREG CR/0660, February 1979, Enhancement of Onsite Emergency Diesel Generator Reliability

1.5.1 Detailed Discussion

The emergency diesel engine starting system was reviewed to the above documents and found to be in basic compliance.

1.6 DIESEL ENGINE LUBRICATION SYSTEM

Those documents reviewed which provide guidance and/or establish design criteria for the diesel engine lubrication system are:

Standard Review Plan

SRP 9.5.7, Rev. 2, July 1981, Emergency Diesel Engine Lubrication System

Reg Guide

Reg Guide 1.9, Rev. 2, December 1979, Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (for comment)

1.6.1 Detailed Discussion

Reg Guide 1.9 Position 7 requires that all safety trips, except the overspeed and differential trips used for onsite standby electric power, be bypassed during accident conditions. Any other trips which may be implemented during accident conditions must use a system of coincident logic utilizing two or more independent measurements. The latter tripping type is used for the low lube oil pressure trip of the diesel generator for North Anna Unit 3.

There is no impact to meet the requirements of the referenced documents.

1.7 DIESEL ENGINE COMBUSTION AIR INTAKE AND EXHAUST SYSTEM

Those documents reviewed which provide guidance and/or establish design criteria for the diesel generator combustion air intake and exhaust system are:

Standard Review Plan

SRP 9.5.8, Rev. 2, July 1981, Emergency Diesel Engine Combustion Air Intake and Exhaust System

Reg Guide

Reg Guide 1.9, Rev. 2, December 1979, Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (for comment)

1.7.1 Detailed Discussion

The design of the emergency diesel engine combustion air intake and exhaust system satisfies the requirements of SRP 9.5.8 by providing independent systems which are protected from extreme natural phenomena and external and internal missiles as well as dust and gasses which could degrade the system.

M. DIESEL GENERATOR BUILDING

SECTION 2

BUILDING IMPACT ASSESSMENT

The details of system changes are in Section 1. The general changes to the structures are as follows.

2.1 TOTAL BUILDING IMPACT DUE TO ALL LICENSING CHANGES

The original diesel generator building was located west of column lines 38 and was large enough to house four units. The new diesel generator building has been moved closer to the present control building and is large enough to hold two diesel generators. The diesel generators will be larger than originally designed. This size increase is because of increased IE loads due to cold shutdown requirements and added safety-related system redundancy to meet seperation and single failure criteria.

The original diesel generators were air cooled. The increased size of the diesel generators facilitate the need to make the diesel generators water cooled. Service water piping will be routed to the diesel building to provide water cooling to the diesel generators. The fuel oil pumphouse has been relocated adjacent to the diesel generator building.

2.2 3-D SEISMIC IMPACTS

The diesel generator building and fuel oil pumphouse would have to be designed to comply with Reg Guides 1.60, 1.61, and 1.92.

N. YARD

SECTION 1

AREA SYSTEMS

Those documents reviewed which provide guidance and/or establish design criteria for systems or components associated with the yard are:

Standard Review Plans (SRP)

SRP Ch 2, Site Characteristics (all), July 1981

SRP 3.4.1, Rev. 2, July 1981, Flood Protection

SRP 3.5.1.6, Rev. 1, July 1981, Aircraft Hazards

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Product Cleanup System

SRP 9.2.5, Rev. 2, July 1981, Ultimate Heat Sink

SRP 9.2.6, Rev. 2, July 1981, Condensate Storage Facilities

Reg Guides

Reg Guide 1.4, Rev. 2, June 1974, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressure Water Reactors

Reg Guide 1.23, Rev. 0, February 1972, Meteorological Programs in Support of Nuclear Power Plants Reg Guide 1.27, Rev. 2, January 1976, Ultimate Heat Sink for Nuclear Power Plants (for comment)

Reg Guide 1.59, Rev. 2, August 1977, Design Basis Floods for Nuclear Power Plants

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado for Nuclear Power Plants

Reg Guide 1.91, Rev. 1, February 1978, Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (for comment)

Reg Guide 1.102, Rev. 1, September 1976, Flood Protection for Nuclear Power Plants

Reg Guide 1.135, Rev. 0, September 1977, Normal Water Level and Discharge at Nuclear Power Plants (for comment)

Reg Guide 1.145, Rev. 0, August 1980, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (for comment)

1.1 QUENCH SPRAY SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the quench spray (QS) system in the yard are:

Standard Review Plan

SRP 6.5.2, Rev. 1, July 1981, Containment Spray as a Fission Product Cleanup System

1.1.1 Detailed Discussion

That portion of the QS system which exists in the yard was reviewed to the above document and system modifications would be required to achieve compliance. A discussion of these modifications follows.

The portion of the QS system which exists in the yard is the QS tank and piping to and from the QS pumps. SRP 6.5.2 requires that the spray system be designed such that the spray solution maintains the highest possible pH and that this requirement be satisfied by a spray pH in the range of 8.5 to 10.5. In order to ensure the proper pH spray solution, a dedicated QS tank would be required. The tank would be maintained at a predetermined chemistry such that the proper spray solution is ensured for all transient conditions. A chemical addition system would be required to keep the QS tank within technical specifications, but its function would not be required during an accident.

1.2 EMERGENCY CORE COOLING SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the emergency core cooling system (ECCS) in the yard are:

Standard Review Plan

SRP 6.3, Rev. 1, July 1981, Emergency Core Cooling System

1.2.1 Detailed Discussion

That portion of the ECCS which exists in the yard was reviewed to the above document and the following modification would be required to achieve full compliance.

That portion of the ECCS system which exists in tne yard is the refueling water storage tank (RWST) and the decay heat suction piping. SRP 6.3 and Reg Guide 1.139 were reviewed with respect to achieving cold safe shutdown using safety-related systems and providing adequate core cooling for the long term for small break LOCAs. This resulted in the anticipated need to have the RWST and ECCS available for long-term cooling.

Redundant ECCS suction piping and missile protection is recommended. This would ensure the availabilty of the RWST for a single active or passive failure for long-term cooling and cold safe shutdown. Missile protection of the RWST neccessitates its relocation due to structural concerns of the piping tunnel below the RWST.

1.3 FIRE PROTECTION SYSTEM

Those documents reviewed which provide guidance and/or establish criteria for the design basis of the fire protection system are:

Standard Review Plans

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

BTP CMEB 9.5-1, Rev.2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

Reg Guides

None

Other Related Documents

10CFR50, Appendix R, September 1981, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979

1.3.1 Detailed Discussion

The review of the fire protection system was completed to the requirements of the above documents. Reg Guide 1.120, Rev. 1, November 1977, Fire Protection Guidelines for Nuclear Power Plants (for comment), has not been reviewed nor an impact established. Due to the nature and number of comments generated during the first public comment period, the guide has been revised extensively and reissued for comment. During the interim, BTP CMEB 9.5-1 is being used for the evaluation of fire protection provisions of operating plants, of plants under construction, and of applications for construction permits and operating licenses.

The previous design of the yard fire protection system, which provided hydrants in strategic locations, is in compliance with the reviewed documents.

No modifications are required.

1.4 AIRCRAFT HAZARDS

Those documents which provide guidance and/or establish design criteria for consideration of aircraft hazards are:

<u>Standard Review Plans</u>

SRP 2.2.1-2.2.2, Rev. 2, July 1981, Identification of Potential Hazards in Site Vicinity

SRP 3.5.1.6, Rev. 1, July 1981, Aircraft Hazards

1.4.1 Detailed Discussion

A review of airspace usage in the plant vicinity should be updated to the guidelines of SRP 2.2.1-2.2.2. A hazards analysis should be conducted using the guidelines of SRP 3.5.1.6 to verify that special design considerations associated with aircraft hazards are not required.

1.5 ULTIMATE HEAT SINK

The document which provides guidelines and/or establishes design criteria for the ultimate heat sink is:

<u>Reg Guide</u>

Reg Guide 1.27, Rev. 2, January 1976, Ultimate Heat Sink for Nuclear Power Plants (for comment)

1.5.1 Detailed Discussion

An analysis of the probable maximum hurricane is required by SRP 2.4.8, Rev. 2, July 1981, Cooling Water Canals and Reservoirs. The study consists of a review of wave forces, riprap stability, overtopping of the embankment, and spray system stress analysis.

Reg Guide 1.27 states that the design basis temperature for equipment associated with the ultimate heat sink should not be exceeded. The service water reservoir temperatures peak at 104-107 F based on the Ford, Bacon, and Davis study. A review of equipment thermal design capability is being performed.

1.6 FLOOD PROTECTION

The document which provides guidance and/or establishes design criteria for the components associated with the site drainage is:

Reg Guide

Reg Guide 1.102, Rev. 1, September 1976, Flood Protection for Nuclear Power Plants

1.6.1 Detailed Discussion

The current revision of this Reg Guide requires a study of the effects of the local probable maximum precipitation in the plant area. As a result of the proposed design change in the elevation of a railroad spur into Unit 3, a means of drainage relief must be provided. The existing Unit 4 circulating water discharge tunnel will be used in conjunction with a new 15 ft wide concrete drainage channel and drop structure.

The channel and drop structure will be located so that no change in the flood protection for Unit 1 or 2 is required.

This concrete structure is approximately 480 ft long, 15 ft wide, and 6 ft deep. It would empty into the Unit 4 circulating water discharge tunnel by means of a concrete drop structure. The concrete drainage channel would have to be designed for 3-D seismic requirements of Reg Guides 1.60, 1.61, and 1.92.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 1

CONTAINMENT ISOLATION

Those documents reviewed which provide guidance and/or establish criteria for containment isolation are:

<u>Standard Review Plans (SRP)</u>

SRP 5.2.5, Rev. 1, July 1981, Reactor Coolant Pressure Boundary Leakage Detection

SRP 6.2.4, Rev. 2, July 1981, Containment Isolation System

SRP 6.2.6, Rev. 2, July 1981, Containment Leakage Testing

<u>Reg Guides</u>

Reg Guide 1.11, Rev. 0, March 1971, Instrument Lines Penetrating Primary Reactor Containment

Reg Guide 1.45, Rev. 0, May 1973, Reactor Coolant Pressure Boundary Leakage Detection Systems

Reg Guide 1.141, Rev. 1 (draft 2), October 1979, Containment Isolation Provisions for Fluid Systems (for comment)

1.1 DETAILED DISCUSSION

The scope of this section includes the isolation of fluid systems which penetrate the containment boundary, and the design and testing requirements for isolation barriers and actuators. North Anna Unit 3 will comply with the above documents.

Reg Guide 1.11 requires instrument lines that are part of the protection system be provided with automatic isolation valves which fail as is, remain open following an accident but are capable of closing in the event of a line failure, and have valve position indication in the control room. To achieve compliance, the three existing air-operated valves in the containment leakage monitoring system which fail open would have to be replaced with three motor-operated Safety Class 2 valves which fail as is and can be operated from the control room.

The four motor-operated valves will be 3/4 in. Due to their additional weight, they will require 3/4 in. tubing instead of the present 3/8 in. tubing.

The following systems meet the provisions of containment isolation regulatory requirements.

Auxiliary Building

Post-LOCA hydrogen recombiner system

Makeup and purification system

Reactor coolant system

Aerated vent and drain system

Containment vacuum system

Fuel pool cooling and purification system

Primary grade water system

Engineered safety feature component cooling water system

Nonsafety component cooling water system

Containment gas and particulate monitoring system

Primary plant gas supply system

Main Steam Valve House

Main steam system

Main feedwater system

Auxiliary feedwater system

Steam generator drain system

Condenser air removal system

Demineralized water system

Instrument and service air system

Safeguards Building

Recirculation spray system

Decay Heat Area

Quench spray system

Decay heat system

Auxiliary Feedwater Pumphouse

Auxiliary feedwater system

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 2

INSERVICE INSPECTION AND TESTING

Those documents reviewed which provide guidance and/or establish criteria for inservice inspection are:

<u>Standard Review Plans (SRP)</u>

SRP 3.9.6, Rev. 2, July 1981, Inservice Testing of Pumps and Valves

SRP 5.2.4, Rev. 1, July 1981, Reactor Coolant Pressure Boundary Inservice Inspection and Testing

SRP 6.6, Rev. 1, July 1981, Inservice Inspection of Class 2 and 3 Components

SRP 14.2, Rev. 2, July 1981, Initial Plant Test Programs - Final Safety Analysis Report

<u>Reg Guides</u>

Reg Guide 1.68, Rev. 2, August 1978, Initial Test Programs for Water-Cooled Reactor Power Plants

Reg Guide 1.147, Rev. 0, February 1981, Inservice Inspection Code Case Acceptability ASME Section XI, Division 1

2.1 DETAILED DISCUSSION

The scope of this section includes the initial test programs and inservice inspection programs that will be used by North Anna Unit 3.

The initial test program for structures, systems, and components whose functions are designated by the General Design Criteria (GDC) of Appendix A to 10CFR50 will comply with the requirements of Reg Guide 1.68.

The initial test program for those structures, systems, and components that are unrelated to functions designated in the GDC will be tested according to their importance to plant reliability.

The inservice inspection requirements for Safety Class 1, 2, and 3 components and their supports are contained in Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code, or equivalent quality standards. The Inservice Inspection Code Cases listed in

Reg Guide 1.147 are limited to those cases applicable to ASME Section XI of the Code which may be applied without requesting specific approval from the NRC.

When an ASME Section XI Code Case is used for Safety Class 1, 2, and 3 components and their supports, it will be reviewed against the latest editions of ASME Section XI Code Cases listed in Reg Guide 1.147. If it is not included, specific approval from the NRC shall be requested. The impact of invoking Inservice Inspection Code Cases will be determined after evaluation of the components and their supports against the latest edition of Reg Guide 1.147.

The inservice testing of pumps and valves that are designated as Class 1, 2, or 3 under Section III of the ASME Boiler and Pressure Vessel Code will comply with the requirements of SRP 3.9.6. ASME XI inservice testing of pumps and valves must comply with the latest code revision that was approved by the NRC and in effect 12 months prior to commercial operation.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 3

FLUID SYSTEM MATERIALS

Those documents reviewed which provide guidance and/or establish criteria for material usage in fluid systems are:

<u>Standard Review Plans (SRP)</u>

SRP 5.2.3, Rev. 2, July 1981, Reactor Coolant Pressure Boundary Materials

SRP 6.1.1, Rev. 2, July 1981, Engineered Safety Features Materials

BTP MTEB 6-1, Rev. 2, July 1981, pH for Emergency Coolant Water for PWRs

SRP 6.1.2, Rev. 2, July 1981, Protective Coating Systems (Paints) - Organic Materials

SRP 10.2.3, Rev. 1, July 1981, Turbine Disk Integrity

SRP 10.3.6, Rev. 2, July 1981, Steam and Feedwater System Materials

<u>Reg Guides</u>

Reg Guide 1.31, Rev. 3, April 1978, Control of Ferrite Content in Stainless Steel Weld Metal

Reg Guide 1.36, Rev. 0, February 1973, Nonmetallic Thermal Insulation for Austenitic Stainless Steel

Reg Guide 1.37, Rev. 0, March 1973, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Reg Guide 1.44, Rev. 0, May 1973, Control of the Use of Sensitized Stainless Steel

Reg Guide 1.50, Rev. 0, May 1973, Control of Preheat Temperature for Welding of Low-Alloy Steel

Reg Guide 1.54, Rev. 0, June 1973, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

Reg Guide 1.71, Rev. 0, December 1973, Welder Qualification for Areas of Limited Accessibility

Reg Guide 1.85, Rev. 15, May 1979, Materials Code Case Acceptability - ASME Section III, Division I

3.1 DETAILED DISCUSSION

All Class 1, 2, 3, and 4 fluid systems have been reviewed with respect to the material usage requirements of the above documents. Several changes regarding material usage would be required to achieve full compliance.

The impact of SRP 6.1.1, par. II.1, and Reg Guide 1.44 on ESF equipment and components currently in storage requires verification that documentation is available to demonstrate that wrought austenitic stainless steel materials with a carbon content greater than 0.035 percent were water quenched. If these materials were not water quenched, they should be tested using a modified ASTM A262 Practice A test which can be performed onsite.

Position C4 requires ESF material subjected to sensitizing temperatures in the range of 800 F to 1,500 F subsequent to solution heat treating L Grade material. Several exceptions are allowed. In the past, material procurements have been conducted in accordance with Position C4. However, IE Bulletins 79-06 and 79-17 have led to the use of low carbon (L Grade) stainless steels for most safety-related systems because L Grade materials exhibit properties which simplify welding operations and reduce the possibility of intergranular stress corrosion cracking.

There is no direct impact associated with material that has already been purchased. Non-L Grade austenitic stainless steel valves and other components that have been procured for use in safety-related systems will meet the testing and welding requirements specified in the Reg Guide 1.44. Non-L Grade austenitic stainless steel piping that has been procured will be used in nonsafety-related systems that meet specific fluid temperature, pH, and flow requirements.

Reg Guide 1.44 has no impact on austenitic stainless steels with a carbon content less than 0.035 percent; nor is there any impact on castings with a carbon content greater than 0.035 percent and ferrite content greater than 5 percent.

Welding and testing of steel and austenitic stainless steel would be in accordance with Reg Guides 1.31, 1.44, 1.50, and 1.71. Insulation and coatings would comply with Reg Guides 1.36 and 1.54 and SRP 6.1.2. Coatings would be qualified to post-LOCA as well as main steam line break environments, as required by SRP 6.1.2.

Additional impacts of ensuring compliance with material usage requirements involve:

1. Reviews of the modifications to the recirculation spray system, quench spray system, makeup and purification system, and decay heat removal system to the requirements of Reg Guide 1.44.

2. Lake water corrosion studies of the component cooling water system, service water system, and circulating water system.

3. Ensuring the cleanliness of system components and controlling contaminants.

4. Ensuring the compatibility of protective coatings inside the reactor containment to the design basis accident environment, as required by Reg Guide 1.54.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 4

SEISMIC AND ENVIRONMENTAL QUALIFICATION, DYNAMIC TESTING, AND ANALYSIS

Those documents reviewed which provide guidance and/or establish criteria for equipment qualification are:

Standard Review Plans

SRP 3.7.1, Rev. 1, July 1981, Seismic Design Parameters

SRP 3.7.2, Rev. 1, July 1981, Seismic System Analysis

SRP 3.7.3, Rev. 1, July 1981, Seismic Subsystem Analysis

SRP 3.9.2, Rev. 1, August 1978, Dynamic Testing and Analysis of Systems, Components, and Equipment

SRP 3.9.3, Rev. 1, July 1981, ASME Code Classes 1, 2, and 3 Components, Component Supports, and Core Support Structures

SRP 3.10, Rev. 1, April 1978, Seismic Qualification of Category I Instrumentation and Electrical Equipment

SRP 3.11, Rev. 1, July 1978, Environmental Qualification of Mechanical and Electrical Equipment

Reg Guides

Reg Guide 1.40, Rev. 0, March 1973, Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

Reg Guide 1.48, Rev. 0, May 1973, Design Limits and Loading Combinations for Seismic Category I Fluid System Components

Reg Guide 1.60, Rev. 1, December 1973, Design Response Spectra for Seismic Design of Nuclear Power Plants

Reg Guide 1.61, Rev. 0, October 1973, Damping Values for Seismic Design of Nuclear Power Plants

Reg Guide 1.63, Rev. 2, July 1978, Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.73, Rev. 0, January 1974, Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Reg Guide 1.89, Rev. 0, November 1974, Environmental Qualification of Electric Equipment Important to Safety for Light-Water-Cooled Nuclear Power Plants

Reg Guide 1.92, Rev. 1, February 1976, Combining Modal Responses and Spatial Components in Seismic Response Analysis

Reg Guide 1.100, Rev. 1, August 1977, Seismic Qualification of Electric Equipment for Nuclear Power Plants

Reg Guide 1.122, Rev. 1, February 1978, Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components

Reg Guide 1.131, Rev. 1 (draft), August 1979, Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants (for comment)

4.1 SEISMIC AND DYNAMIC QUALIFICATION

North Anna Unit 3 seismic system components still to be purchased would have to comply with the requirements of SRP 3.9.3.

Existing components are qualified under ASME III Code Cases 1607 for Class 2 and 3 vessels, 1635 for Class 2 and 3 valves, and 1636 for Class 2 and 3 pumps. All components still to be purchased would have to be qualified under ASME III subsections NB, NC, ND, and NF.

Component operability is assured by satisfying the requirements of various programs. Safety-related valves and pumps are qualified by testing, i.e., hydrostatic tests per ASME III, seal leakage tests, performance tests, etc; or by analysis.

North Anna Unit 3 system seismic analysis would have to comply with Reg Guides 1.60, 1.61, 1.92, and 1.122 for all Seismic Category I components. All Category I components would have to be reviewed for 3-D seismic adequacy and requalified or repurchased as required.

The use of 3-D amplified response spectra for all Category I piping in the SWEC scope of work would entail reanalyzing about 100 of the 650 stress summaries and reviewing the associated preliminary pipe support designs, break locations, jet and rupture restraint designs, containment penetration and nozzle loads, and insert and overlay pads on the liner.

4.2 EQUIPMENT QUALIFICATION

Per SRP 3.10, operability of mechanical and electrical equipment would have to be assured under normal, accident, and seismic loadings for which the operation of the component is required.

Design adequacy would have to be re-reviewed against the new 3-D criteria and ARS curves generated since the purchase. The use of single axis, single frequency test input motion will be justified and documented or the equipment would be requalified/repurchased to the multiaxis multifrequency criteria of IEEE 344-1975.

Seismic and dynamic input motion used for seismic qualification will be described by Amplified Response Spectrum (ARS) for each building and elevation. Actual test motions will cover, at a minimum, the required response spectrum (RRS) over the critical frequency ranges. RRS will equal the ARS. Damping values will be in accordance with Reg Guide 1.61. Test conditions will consider the effects of dynamic coupling, actual monitoring methods, and orientation. Vibratory devices to simulate seismic and dynamic motions would be used in situ and demonstrated as valid when other methods are impractical. Prototype testing will be used when possible. The seismic and dynamic testing portion of the overall qualification would be performed in its proper sequence, as indicated in Section 6 of IEEE 323-1974.

Static testing of pump and valve assemblies would be used when dynamic testing is not possible due to factors such as the component's size and/or weight. Sufficient conservatism would be applied on end loadings to simulated postulated event loads and dynamic amplification effects.

When complete testing is not practical, a combination of test and analysis methods would be utilized. Complex active devices which are a part of a complete assembly, such as pump motors, valve operators, solenoids, and other appurtenances, would be tested for operability. Remaining parts of the assemblies would be qualified by analysis using methods outlined in Section II.1.1.14.b of SRP 3.10.

4.3 CLASS 1 PUMPS

Class 1 pumps would be designed and analyzed according to ASME Section III, Subsection NB 3400, as endorsed by Reg Guide 1.48. In addition to these tests, the safety-related active pumps are qualified for operability during a safe shutdown earthquake (SSE) condition by assuring that the pump is not damaged during the seismic event and the pump continues operating when subjected to the SSE loads. The pump motor and vital auxiliary equipment are seismically qualified by meeting the requirements of IEEE 344.

4.4 VALVE OPERABILITY PROGRAM

SRP 3.10 requires qualification tests accompanied by analysis for active valves to ensure operability during a seismic event.

Valves without extended structures are proven seismically adequate by analysis of piping system adequacy. If valves with operators have significant extended structures, and if these structures are essential to maintaining the pressure boundary integrity, analysis is performed based upon static forces resulting from equivalent earthquake accelerations acting at the centers of gravity of the extended masses.

Representative valves of each design type with extended structures are tested for verification of operability during a simulated seismic event.

4.5 VERIFICATION OF SEISMIC AND DYNAMIC QUALIFICATION

Existing equipment meets the seismic and dynamic requirements which were in effect at the time of their purchase. Review, re-evaluation, and requalification of this equipment would be required to upgrade to the criteria of subsection II.1 of SRP 3.10. Existing electrical equipment would meet, at a minimum, the requirement of IEEE 344-1971, with operability demonstrated and documented while considering the effects of multimode response, multiaxis excitation, and multifrequency input excitations.

An equipment qualification file would be maintained and would contain the information required by subsections II.3, II.4, and II.5. of SRP 3.10.

4.5.1 Impact

Implementation of a program to meet the acceptance criteria of SRP 3.10, as delineated above, would result in the following significant impact.

All active valves with extended structures would be grouped for the purpose of qualifying several valves on the basis of testing of a representative member of the group. It is anticipated that 20 to 30 individual valves would require testing.

4.6 DYNAMIC TESTING AND ANALYSIS

North Anna Unit 3 would be required comply with the requirements of SRP 3.9.2. The scope of the following discussion is limited to the dynamic testing and analysis of systems, components, and equipment as addressed in SRP 3.9.2, excluding reactor internals and NSSS components.

A preoperational vibration test program on ASME Code Classes 1, 2, and 3 piping systems within balance of plant (BOP) scope would be conducted under simulated transients within the normal and upset operating modes of the systems. Selected locations on the piping systems are subjected to visual inspection and instrumented measurements are performed, if needed, during the following tests:

1. Start and stop reactor coolant pumps with associated operation of valves (closures/openings) in primary reactor coolant piping systems. Similar testing would also be performed on the main steam and feedwater systems.

2. Start and stop decay heat removal (DHR) pumps with normal operation (closures/openings) of the associated valves in DHR piping systems.

3. Operation of high pressure injection piping system and makeup system.

4. Operation of pressurizer relief valves and associated discharge piping system.

5. Start and stop auxiliary feedwater pump with normal operation (closures/openings) of the associated valves in the auxiliary feedwater piping system.

During the preoperational and initial startup test program, if excessive vibration is visually observed and confirmed by instrumentation on a BOP ASME Code Classes 1, 2, or 3 piping system, corrective support systems would be designed and installed and the effect of the modification would be incorporated in the pipe stress analysis. If instrumented testing is required, the selection of measurement stations in the test program is to ensure the adequacy of qualifying the pipe systems, and the measurements of dynamic piping responses would be converted to a moment loading and verified to be within allowable code limits when combined with other appropriate operational loadings.

For ASME Code Classes 1, 2, and 3 components and piping systems, design and supervision of the tests, definition of acceptance criteria, evaluations of test results, and any changes in the piping systems necessary to ensure that the piping is adequately designed and supported, would be performed as required by Section III of the ASME Code.

The methods and procedures used in the design and qualification of Seismic Category I mechanical equipment within the BOP scope shall meet the criteria of SRP 3.9.2. Loading combinations include operating as well as earthquake loadings for consideration by testing and/or analytical methods.

All Seismic Category I equipment within BOP scope would be shown to have structural integrity during all plant conditions by analysis satisfying the stress criteria applicable to the particular piece of equipment or by a test showing that the equipment retains its structural integrity under the simulated test environment.

When equipment can be characterized as relatively simple and when acceptability can be demonstrated by stress, strain, or deflection calculations, static analysis is performed. The design and associated analysis shall account for the relative motion between all points of support.

Compliance with SRP 3.9.2 will not impact the present scope of work.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 5

PROTECTION FROM PIPE BREAK

The documents reviewed which provide guidance and/or establish criteria for the design basis with respect to pipe break are:

Standard Review Plans (SRP)

SRP 3.6.1, Rev. 1, July 1981, Plant Design for Protection against Postulated Piping Failures in Fluid Systems outside Containment

BTP ASB 3-1, Rev. 1, July 1981, Protection against Postulated Piping Failures in Fluid Systems outside Containment

SRP 3.6.2, Rev. 1, July 1981, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

BTP MEB 3-1, Rev. 1, July 1981, Postulated Rupture Locations in Fluid System Piping inside and outside Containment

5.1 DETAILED DISCUSSION

Criteria governing the treatment of pipe breaks inside the containment would be required to conform to SRP 3.6.2, including BTP MEB 3-1. Reg Guide 1.46, Rev. 0, May 1973, Protection Against Pipe Whip inside Containment, will not be cited because it is no longer referenced in SRP 3.6.2. Criteria governing the treatment of pipe breaks outside the containment would be required to conform to BTP ASB 3-1 of SRP 3.6.1 and BTP MEB 3-1 of SRP 3.6.2, with the following exception. Jet impingement on essential equipment in break exclusion areas identified in SRP 3.6.1, par. B121, will not be evaluated.

The direct consequences of a pipe rupture, such as unit trip, would be taken into account as well as the consequences of an assured single failure and loss of offsite power.

When protective measures for essential systems are required, one of the following design methods would be employed based on cost effectiveness and level of protection.

1. Maximize separation between high energy piping and essential systems

2. Provide enclosures capable of withstanding pipe rupture effects

3. Provide limited separation such that the break impact on the essential system is reduced to an acceptable level

4. Provide pipe rupture restraints

5. Redesign target to withstand rupture loading

6. Provide jet impingement barriers at the target or source

7. Establish an inservice inspection program in lieu of break postulation (subject to NRC approval)

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 6

PROTECTION FROM INTERNALLY GENERATED MISSILES

The documents reviewed which provide guidance and/or establish criteria for the design basis with respect to internally generated missiles are:

Standard Review Plans (SRP)

SRP 3.5.1.1, Rev. 2, July 1981, Internally Generated Missiles (outside containment)

SRP 3.5.1.2, Rev. 2, July 1981, Internally Generated Missiles (inside containment)

SRP 3.5.1.3, Rev. 2, July 1981, Turbine Missiles

Reg Guide

Reg Guide 1.115, Rev. 1, July 1977, Protection Against Low-Trajectory Turbine Missiles

6.1 DETAILED DISCUSSION

North Anna Unit 3 would be required to comply with the requirements of SRP 3.5.1.1 and SRP 3.5.1.2. An analysis of internally generated missiles would be conducted to demonstrate compliance with the design acceptance criteria presented in the SRPs. Missiles can be generated from pressurized components or high speed rotating equipment. All safety-related structures, systems, and components in the path of the postulated missile are considered. The documentation would identify each design missile and its trajectory, velocity, and energy. Each design basis target would be documented to show the basis for acceptance.

The North Anna Unit 3 turbine missile generation strike and damage probabilities would be calculated based on the guidelines contained in SRP 3.5.1.3 and Reg Guide 1.115. The following acceptance criteria would be advanced in the turbine missile analysis.

1. The probability of occurrence of potential radiation exposures in excess of 10CFR100 limits is no greater than 10^{-6} per year.

2. The probability of damage summed over all essential systems located within the low trajectory missile strike zone will be held to a sufficiently low value so that its contribution, when combined with the missile

ejection and strike damage probabilities, will result in an overall probability of exposure exceeding 10CFR100 guidelines by no more than 10^{-6} per year. To simplify the evaluation, probabilities are summed over cubicles containing essential systems.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 7

LIGHTING SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the lighting system are:

<u>Standard Review Plan (SRP)</u>

SRP 9.5.3, Rev. 2, July 1981, Lighting System

<u>Reg Guides</u>

None

7.1 DETAILED DISCUSSION

The North Anna Unit 3 lighting system would be required to provide adequate station lighting during normal and transient plant conditions, as required by SRP 9.5.3

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 8

COMMUNICATIONS SYSTEM

The documents reviewed which provide guidance and/or establish criteria for the design basis of the communications system are:

Standard Review Plan

SRP 9.5.2, Rev. 2, July 1981, Communications System

Reg Guide

Reg Guide 8.5, Rev. 1, March 1981, Criticality and Other Interior Evacuation Signals

Other Documents

NRC IE Bulletin 79-18, Audibility Problems Encountered on Evacuation

8.1 DETAILED DISCUSSION

The North Anna Unit 3 communications system would be required to provide effective intraplant and plant-to-offsite communications during normal and transient plant conditions, including loss of offsite power, as required by SRP 9.5.2 and Reg Guide 8.5. The need for visual alarm systems will be considered when noise level studies are conducted as required by SRP 9.5.2 and NRC IE Bulletin 79-18.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 9

ELECTRICAL SYSTEMS

Those documents reviewed which provide guidance and/or establish criteria for electrical systems are:

Standard Review Plans (SRP)

SRP 8.1, Rev. 2, July 1981, Electric Power - Introduction

SRP 8.2, Rev. 2, July 1981, Offsite Power System

SRP 8.3.1, Rev. 2, July 1981, A-C Power Systems (onsite)

SRP 8.3.2, Rev. 2, July 1981, D-C Power Systems (onsite)

Reg Guides

Reg Guide 1.6, Rev. 0, March 1971, Independence between Redundant Standby (onsite) Power Sources and between Their Distribution Systems (Safety Guide 6)

Reg Guide 1.32, Rev. 2, February 1977, Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Reg Guide 1.41, Rev. 0, March 1973, Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

Reg Guide 1.81, Rev. 1, January 1975, Shared Emergency and Shutdown Electric Systems for Multi-Unit Power Plants

Reg Guide 1.93, Rev. 0, December 1974, Availability of Electric Power Sources

Reg Guide 1.106, Rev. 1, March 1977, Thermal Overload Protection for Electric Motors on Motor-Operated Valves

Reg Guide 1.108, Rev. 1, August 1977, Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

Reg Guide 1.118, Rev. 2, June 1978, Periodic Testing of Electric Power and Protection Systems

Reg Guide 1.128, Rev. 1, October 1978, Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

Reg Guide 1.129, Rev. 1, February 1978, Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

Other Related Documents

BTP PSB 1, Rev. 0, July 1981, Adequacy of Station Shutdown Electric Distribution System Voltages

BTB PSB 2, Rev. 0, July 1981, Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status

9.1 DETAILED DISCUSSION

With one exception, the offsite power system interface within the SWEC scope is in compliance with the referenced documents. To comply with Reg Guide 1.93, a voltmeter (0-5.25 kV scale) would be installed in the vicinity of circuit breaker 6 bus A4 for monitoring and indicating the status of the preferred power system. An undervoltage alarm would indicate in the control room if any change in the preferred power system would prevent it from performing its intended function.

The onsite ac and dc power systems are in compliance with the referenced documents and the following impact was identified. A review of the adverse effects of sustained low voltage conditions on Class 1E loads when the Class 1E buses are connected to offsite power should be conducted. This review, addressing the concerns of BTP PSB 1, would determine the adequacy of distribution system low voltage detection. The voltage protection logic would be reviewed to ensure protection against adverse effects on the Class 1E systems, e.g., spurious separation from offsite power due to normal motor starting transients.

Separation of safety-related electric systems is addressed in Reg Guide 1.75.

The safety-related diesel-generator units have not been purchased for North Anna Unit 3. When these units are purchased, they would be required to comply with the applicable regulatory documents. These units will be larger than the units originally specified because of increases in Class 1E electrical loads.

The existing unit substation design calls for two 480 V stub-buses to feed critical non-Class 1E loads from emergency load centers. These stub-buses are connected to the emergency buses by electrically-operated Class 1E breakers that would be tripped and locked out upon receiving an engineered safeguards activation signal generated within its division. Therefore, they qualify as an isolation device.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 10

HAZARDS ANALYSIS

Those documents reviewed which provide guidance and/or establish criteria for hazards evaluation are:

- Standard Review Plans (SRP)

SRP 3.5.1.1, Rev. 2, July 1981, Internally Generated Missiles (outside containment)

SRP 3.5.1.2, Rev. 2, July 1981, Internally Generated Missiles (inside containment)

SRP 3.5.1.3, Rev. 2, July 1981, Turbine Missiles

SRP 3.5.3, Rev. 1, July 1981, Barrier Design Procedures

SRP 3.6.1, Rev. 1, July 1981, Plant Design for Protection against Postulated Piping Failures in Fluid Systems outside Containment

SRP 3.6.2, Rev. 1, July 1981, Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

SRP 3.9.3, Rev. 1, July 1981, ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

SRP 3.11, Rev. 2, July 1981, Environmental Qualification of Mechanical and Electrical Equipment

SRP 9.5.1, Rev. 3, July 1981, Fire Protection System

Reg Guides

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.62, Rev. 0, October 1973, Manual Initiation of Protective Action

Reg Guide 1.75, Rev. 2, September 1978, Physical Independence of Electric Systems

Reg Guide 1.115, Rev. 1, July 1977, Protection Against Low-Trajectory Turbine Missiles

10.1 SCOPE

The scope of this activity would include the evaluation of the interaction between each postulated hazard and each piece of equipment essential to the safety of the plant. It is an administrative program which would use equipment arrangement and analysis data developed by the project disciplines to account for each hazard/equipment interaction. It would document the disposition of each interaction and make substantiating calculations traceable. Evaluation would be carried out on a zone by zone basis in each building housing such equipment. Criteria and procedures to be followed are presented in Hazards Evaluation and Documentation Procedure, Rev. 0, December 12, 1981. Each interaction would be documented on marked up arrangement plans where possible and in a computerized Hazards Tracking List. The evaluation and documentation of all interactions, including those that would require resolution of potential hazards, would assure that the plant both can attain and maintain safe shutdown while keeping offsite dosage within 10CFR100 limits.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 11

SUBCOMPARTMENT PRESSURIZATION ANALYSIS

The documents reviewed which provide guidance and/or establish criteria for the design basis with respect to subcompartment pressurization are:

<u>Standard Review Plan (SRP)</u>

SRP 6.2.1.2, Rev. 2, July 1981, Subcompartment Analysis

<u>Other Related Documents</u>

NUREG/CR 1199, December 1979, Subcompartment Analysis Procedures

NUREG 0609, January 1981, Asymmetric Blowdown Loads on PWR Primary Systems

11.1 DETAILED DISCUSSION

Subcompartment analysis would be required by SRP 6.2.1.2. With the issuance of NUREG 0800 in July 1981, SRP 6.2.1.2 was revised to require the utilization of the guidelines of NUREG 0609, January 1981, Section 3.2. This includes consideration of spatial pressure variation within the subcompartment for use on calculating the transient forces and moments acting on components.

North Anna Unit 3 would be required to comply with the provisions of SRP 6.2.1.2.

NUREG 0609, as invoked by SRP 6.2.1.2, is the culmination of a generic task action plan initiated by the NRC to study the phenomena of asymmetric pressure loads resulting from postulated pipe ruptures in the primary coolant system.

NUREG 0609 very specifically requires performance of subcompartment analysis in such a way as to assess both the asymmetric pressure effects (loads) induced by a high energy line rupture immediately adjacent to the structure or component and by a pipe rupture which is not immediately adjacent to the component/structure or where the worst case loading results from an overturning moment created by loads away from the break.

In order to analyze pipe rupture effects both adjacent to and removed from the components/structures of interest, the subcompartment models must possess sufficient detail to predict the near and far field effects of a pipe rupture.

In developing the pressure-time histories, it is typically necessary to perform several distinct analyses utilizing different critical flow models in order to generate conservative pressure-time histories for near and far effects of pipe ruptures.

O. GENERAL SYSTEMS AND SYSTEM-RELATED ITEMS

SECTION 12

QUALITY ASSURANCE

Those documents reviewed which provide guidance and/or establish criteria for quality assurance (QA) are:

<u>Standard Review Plans (SRP)</u>

SRP 17.1, Rev. 2, July 1981, Quality Assurance During the Design and Construction Phase

SRP 17.2, Rev. 2, July 1981, Quality Assurance During the Operation Phase

BTP CMEB 9.5-1, Rev. 2, July 1981, Guidelines for Fire Protection for Nuclear Power Plants

<u>Reg Guides</u>

Reg Guide 1.8, Rev. 1, May 1977, Personnel Selection and Training

Reg Guide 1.26, Rev. 3, February 1976, Quality Group Classification, and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Reg Guide 1.28, Rev. 2, February 1979, Quality Assurance Program Requirements (design and construction)

Reg Guide 1.29, Rev. 3, September 1978, Seismic Design Classification

Reg Guide 1.30, Rev. 0, August 1972, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

Reg Guide 1.37, Rev. 0, March 1973, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Reg Guide 1.38, Rev. 2, May 1977, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

Reg Guide 1.39, Rev. 2, September 1977, Housekeeping Requirements for Water-Cooled Nuclear Power Plants

Reg Guide 1.58, Rev. 1, September 1980, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

Reg Guide 1.64, Rev. 2, June 1976, Quality Assurance Requirements for the Design of Nuclear Power Plants

Reg Guide 1.74, Rev. 0, February 1974, Quality Assurance Terms and Definitions

Reg Guide 1.88, Rev. 2, October 1976, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records

Reg Guide 1.94, Rev. 1, April 1976, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

Reg Guide 1.116, Rev. O-R, May 1977, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

Reg Guide 1.123, Rev. 1, July 1977, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

Reg Guide 1.144, Rev. 1, September 1980, Auditing of Quality Assurance Programs for Nuclear Power Plants

Reg Guide 1.146, Rev. 0, August 1980, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

12.1 DETAILED DISCUSSION

The QA program applicable to the design and construction of North Anna Unit 3 is contained in the Virginia Electric and Power Company report on North Anna Unit 3. This report, when approved by the NRC, would replace the current program description in Appendix A of the North Anna Unit 3 PSAR.

The major changes reflected in this report, as compared to Appendix A of the PSAR, are as follows

- Updating to reflect the current VEPCO organizational structure and responsibilities.
- Changes to reflect the role of VEPCO as construction manager for North Anna Unit 3.
- Changes to reflect the revision of SWEC's Scope of Work. SWEC has responsibily for design and selected procurement.

- Adoption of the SWEC topical report, SWSQAP 1-74A, and the B&W topical report, BAW-10096A, which describe the QA programs of the architect/engineer and nuclear steam system supplier, respectively.

- Update of commitments to regulatory guidance.

APPENDIX 1

SECTION 2 RELATED DOCUMENTS

Except when noted in Sections 2.1 and 2.2 of the individual building impact statements, the Reg Guides and Standard Review Plans (SRP) listed below were reviewed and no major structural impacts were identified:

Flood Protection

SRP 2.4.2, Rev. 2, July 1981, Floods

SRP 2.4.10, Rev. 2, July 1981, Flood Protection Requirements

SRP 3.4.1, Rev. 2, July 1981, Flood Protection

SRP 3.4.2, Rev. 1, July 1981, Analysis Procedures

Reg Guide 1.59, Rev. 2, August 1977, Design Basis Floods for Nuclear Power Plants

Reg Guide 1.102, Rev. 1, September 1976, Flood Protection for Nuclear Power Plants

Wind Loadings

SRP 3.3.1, Rev. 2, July 1981, Wind Loadings

Tornado and Hurricane Loading and Protection

SRP 3.3.2, Rev. 0, November 1975, Tornado Loading

Reg Guide 1.76, Rev. 0, April 1974, Design Basis Tornado

Reg Guide 1.117, Rev. 1, April 1978, Tornado Design Classification

Protection from Externally Generated Missiles

SRP 3.5.1.3, Rev. 2, July 1981, Turbine Missiles

SRP 3.5.1.4, Rev. 2, July 1981, Missiles Generated by Natural Phenomena

SRP 3.5.1.5, Rev. 1, July 1981, Site Proximity Missiles (except aircraft)

SRP 3.5.1.6, Rev. 1, July 1981, Aircraft Hazards

SRP 3.5.2, Rev. 2, July 1981, Structures, Systems, and Components to Be Protected from Externally Generated Missiles

SRP 3.5.3, Rev. 1, July 1981, Barrier Design Procedures

Reg Guide 1.115, Rev. 1, July 1977, Protection Against Low-Trajectory Turbine Missiles

Building Seismic Analysis

SRP 2.5.2, Rev. 1, July 1981, Vibratory Ground Motion

Building Materials and Construction

SRP 2.5.4, Rev. 2, July 1981, Stability of Subsurface Materials and Foundations

SRP 3.7.4, Rev. 1, July 1981, Seismic Instrumentation

SRP 3.8.1, Rev. 1, July 1981, Concrete Containment

SRP 3.8.3, Rev. 1, Concrete and Steel Internal Structures of Steel or Concrete Containments

SRP 3.8.4, Rev. 1, July 1981, Other Seismic Category I Structures

SRP 3.8.5, Rev. 1, July 1981, Foundations

SRP 3.9.1, Rev. 2, July 1981, Special Topics for Mechanical Components

SRP 12.3, Rev. 2, July 1981, Radiation Protection Design Features

Reg Guide 1.10, withdrawn, July 1981, Mechanical (cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Reg Guide 1.15, withdrawn, July 1981, Testing of Reinforcing Bars for Category I Concrete Structures

Reg Guide 1.55, withdrawn, July 1981, Concrete Placement in Category I Structures

Reg Guide 1.69, Rev. 0, December 1973, Concrete Radiation Shields for Nuclear Power Plants

Reg Guide 1.94, Rev. 2 (draft), September 1979, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel, Soils, and Foundations during the Construction Phase of Nuclear Power Plants

Reg Guide 1.132, Rev. 1, March 1979, Site Investigations for Foundations of Nuclear Power Plants

Reg Guide 1.142, Rev. 1, October 1981, Safety-Related Concrete Structures for Nuclear Power Plants (other than reactor vessels and containments) (for comment)

APPENDIX 2

RECENT REVISIONS TO STRUCTURAL DESIGN CRITERIA

The NRC has issued regulatory documents which would affect the present structural design. These documents are:

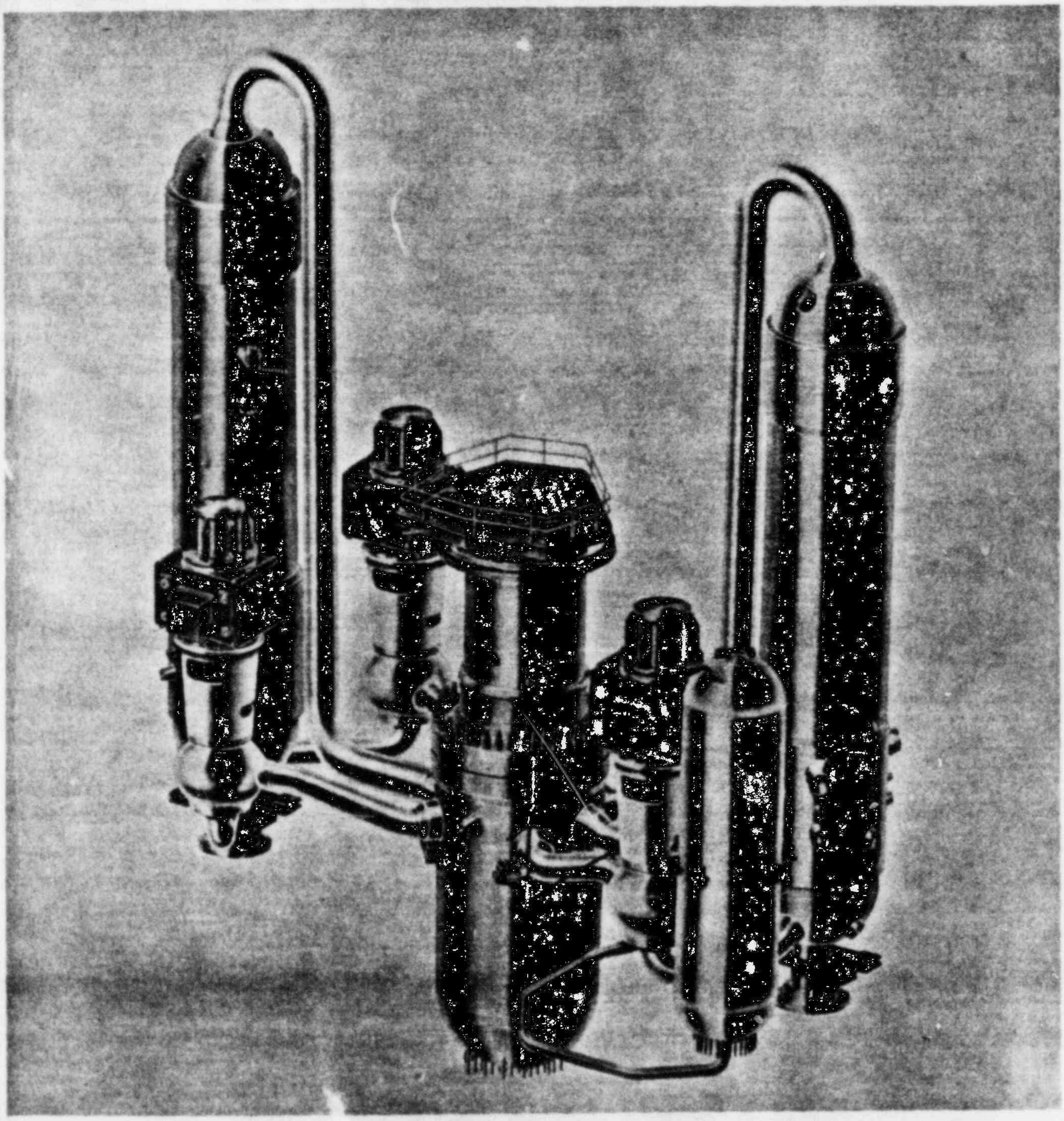
SRP 3.8.4, Rev. 1, July 1981, Other Seismic Category I Structures

Reg Guide 1.136, Rev. 2, June 1981, Materials, Construction, and Testing of Concrete Containment

In order to comply with the provisions of SRP 3.8.4, the calculation procedures and the structural design criteria would have to be revised. The Design Report would have to be written when design is complete.

To meet the requirements of Reg Guide 1.136, the concrete specifications for the reactor containment would have to be revised.

VEPCO
North Anna Power Station
Unit 3



Babcock & Wilcox
a McDermott company